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Report of the U.S. Nuclear Regulatory Commission Piping Review Committee

Evaluation of Potential for Pipe Breaks

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Prepared by
The Pipe Break Task Group

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

PIPE BREAK TASK GROUP MEMBERS

R. W. Klecker, Chairman

S. H. Bush
S. H. Hou
J. Strosnider
K. R. Wichman

PIPE BREAK TASK GROUP CONSULTANTS

C. K. Chou
R. Gamble
G. Holman
G. Wilkowski

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FOREWORD

The Executive Director for Operations of the U. S. Nuclear Regulatory Commission (NRC) requested that a comprehensive review be made of NRC requirements in the area of nuclear power plant piping. In response to this request an NRC Piping Review Committee was formed. The activities of this review committee were divided into four tasks handled by appropriate task groups, namely:

- Pipe Crack Task Group
- Seismic Design Task Group
- Pipe Break Task Group
- Dynamic Load/Load Combination Task Group.

Each task group will prepare a report appropriate to its scope. In addition, the Piping Review Committee will prepare an overview document rationalizing areas of overlap between the task groups. This will be released as a separate report.

Because of the nature of the current intergranular stress corrosion cracking (IGSCC) problems in boiling water reactors (BWRs), the Pipe Crack Task Group was on an accelerated schedule. This report was due in March-April 1984, while the other task groups are aiming for August-September 1984. The Review Committee should complete its activities prior to the end of 1984.

The project titles of the five volumes that make up Report of the U.S. Nuclear Regulatory Commission Piping Review Committee are:

Volume I - Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants

Volume II - Evaluation of Seismic Designs

Volume III - Evaluation of Potential for Pipe Breaks

Volume IV - Evaluation of Other Dynamic Loads and Load Combinations

Volume V - Summary - Piping Review Committee Conclusions and Recommendations.

This report deals with the potential for pipe breaks and recommends modifications to the existing position.

ACKNOWLEDGMENTS

The members of the Task Group on Pipe Break wish to express their appreciation for the active support given them by members of the staff of Battelle Memorial Institute-Columbus during the preparation of NUREG-1061 Volume III, "Evaluations of Potential for Pipe Breaks".

Specifically, we wish to thank Gery Wilkowski who made all arrangements in addition to serving as a consultant; to Louisa Ronan, his secretary; Yashoda N. Singh, for technical editing; and Sherry Galford and Norma Hunter for word processing.

Appendix D lists the names of members or consultants to the Task Group. In addition, the Task Group met with members of industry, specifically, the Atomic Industrial Forum.

EXECUTIVE SUMMARY

The Executive Director for Operations (EDO) in establishing the Piping Review Committee concurred in its overall scope that included an evaluation of the potential for pipe breaks. The Pipe Break Task Group has responded to this directive.

This report summarizes a review of regulatory documents and contains the Task Group's recommendations for application of the leak-before-break (LBB) approach to the NRC licensing process. The LBB approach means the application of fracture mechanics technology to demonstrate that high energy fluid piping is very unlikely to experience double-ended ruptures or their equivalent as longitudinal or diagonal splits.

The Task Group's recommendations and discussion are founded on current and ongoing NRC staff actions as presented in Section 3.0 of this report. Additional more detailed comments and discussion are presented in Section 5.0 and in Appendices A and B.

The obvious issues are the reexamination of the large pipe break criteria and the implications of any changes in the criteria as they influence items such as jet loads and pipe whip. The issues have been considered and the Task Group makes the following recommendations:

GENERAL RECOMMENDATIONS

(1) A caveat on the use of leak-before-break (LBB) instead of double-ended guillotine break (DEGB) is the absence of excessive loads or cracking mechanisms that could adversely affect the accurate evaluation of flaws and loads. Specific examples include water hammer and water slugging, other large dynamic loads, intergranular stress corrosion cracking (IGSCC), and fatigue.

(2) There should be no change in design bases for systems such as containment, emergency core cooling system (ECCS), component and piping supports, etc., at least in the near future. The DEGB or its equivalent should be retained as a design basis for such systems.

(3) Leak detection systems in existing nuclear plants should be examined on a case-by-case basis to ensure that suitable detection margins exist so that the margin of detection for the largest postulated leakage size crack used in the fracture mechanics analyses is greater than a factor of ten on unidentified leakage. Licensees and applicants have the option of requesting a decrease in leakage margin provided they can confirm that their leakage detection systems are sufficiently reliable, redundant, diverse, and sensitive.

(4) The elimination of the DEGB at terminal ends of large primary pipes in pressurized water reactors (PWRs) and the control of the maximum flaw length in piping in general should permit an elimination of existing restraints or removal of restraints as a design requirement. Consequently, asymmetric reactor pressure vessel (RPV) loads, jet impingement loads, and reactor cavity over-pressurization that result from a postulated DEGB need not be considered.

(5) Arbitrary intermediate pipe breaks should be eliminated as a design basis requirement.

(6) Necessary changes should be made to documents such as Regulatory Guides, Standard Review Plan, and Generic Issues to facilitate the use of fracture mechanics technology in the licensing process.

Codes and Standards (e.g., ASME III and XI) may require changes. Such changes should be presented to the appropriate code or standards body for consideration.

Expedited efforts should be applied to revising some existing regulations via the rulemaking process. An obvious example is General Design Criterion (GDC) 4. These efforts should be given priority over the revision of guidance documents.

(7) In boiling water reactors (BWRs) where piping systems of safety significance have been replaced with a material resistant to IGSCC, such as 316 NG, the DEGB should not be a design criterion.

RECOMMENDATIONS FOR APPLICATION OF THE LBB APPROACH

To provide guidance to potential users of the LBB approach, each step of the process required to develop the requisite technical justification for a LBB submittal is described in general terms below. A detailed description of the acceptance criteria that should be used by the staff for evaluation of each submittal is presented in Section 5.0.

(a) Provide a discussion to support the conclusion that this piping run or system does not fall within the limitations delineated in Section 5.1.

(b) Specify the type and magnitude of the loads applied (forces, bending and torsional moments), their source(s) and method of combination. Identify the location(s) at which the highest stresses coincident with poorest material properties occur for base materials, weldments, and safe-ends.

(c) Identify the types of materials and materials specifications used for base metal, weldments and safe-ends, and provide the materials properties including appropriate toughness and tensile data, long-term effects such as thermal aging and other limitations.

(d) Postulate a flaw at the location(s) specified in (b) above that would be permitted by the acceptance criteria of Section XI of the ASME Boiler & Pressure Vessel Code. Demonstrate by fatigue crack growth analysis for Code Class 1 piping that the crack will not grow significantly during service.

(e) Postulate a throughwall flaw at the location(s) specified in (b); above. The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak detection capability when the pipes are subjected to normal operating loads. If auxiliary leak detection systems are relied on, they should be described.

(f) For geometrically complex lines or systems, performance of a system evaluation should be considered.

(g) Assume that a safe shutdown earthquake (SSE) occurs prior to detection of the leak to demonstrate that the postulated leakage flaw is stable under normal operating plus SSE loads for a long period of time; that is, crack growth if any is minimal during an earthquake.

(h) Determine flaw size margin by comparing the selected leakage size flaw (Item e) to critical size crack. Using normal plus SSE loads, demonstrate that there is a margin of at least 2 between the leakage size flaw and the critical size crack to account for the uncertainties inherent in the analyses and leak detection capability.

(i) Determine margin in terms of applied loads by a crack stability analysis. Demonstrate that the leakage-size cracks will not experience unstable crack growth even if larger loads (at least the $\sqrt{2}$ times the normal plus SSE loads) are applied. Demonstrate that crack growth is stable and the final crack size is limited such that a double-ended pipe break will not occur.

(j) The piping materials toughness (J-R curves) and tensile (stress-strain curves) properties should be determined at temperatures near the upper range of normal plant operation. The test data should demonstrate ductile behavior at these temperatures.

(k) Ideally the J-R curves should be obtained using specimens whose thickness is equal or greater than that of the pipe wall. The specimen should be large enough to provide crack extensions up to an amount consistent with J/T condition determined by analysis for the application. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques may be used if appropriate as described in Section A2.4.3 (Appendix A).

(l) The stress-strain curves should be obtained over the range from the proportional limit to maximum load.

(m) Ideally, the materials tests should be conducted using archival material for the pipe being evaluated. If archival material is not available, tests should be conducted using specimens from three heats of material having the same material specification. Test material should include base and weld metals.

(n) At least two stress-strain curves and two J-resistance curves should be developed for each of a minimum of three heats of materials having the same material specifications and thermal and fabrication histories as the in-service piping material. If the data are being developed from an archival heat of material, a minimum of three stress-strain curves and three J-resistance curves from that one heat of material is sufficient. The tests should be conducted at temperatures near the upper range of normal plant operation (e.g., 550 F). Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. One J-R curve and one stress-strain curve for one base metal and weld metal are considered adequate to determine temperature dependence.

(o) As indicated in Section 5.9.1 there are certain limitations that currently preclude generic use of limit load analyses to evaluate leak-before-break conditions for eliminating pipe restraints. However, the Task Group believes that limit-load analysis can be used to demonstrate acceptable leak-before-break margins for the application, provided the limit moment is greater than the applied (normal operation plus safe shutdown earthquake (SSE)) moment at any location in the pipe run by a factor of at least three. Limit moment should be determined from Eq. (A-19) in Appendix A where the flow stress is determined from ASME Code minimum properties. Data obtained from future tests (see Section 10.0) may provide information that would allow less restrictive use of limit-load analyses for justifying elimination of pipe restraints.

The preceding description of the steps in performing a LBB analysis assumes that circumferentially oriented postulated cracks are limiting. If this is not the case, the analyses described in the above steps should also include the postulation of axial cracks and/or elbow cracks.

DISCUSSION OF ANALYTIC METHODS

- In an attempt to benchmark various J computational methods the Task Group compared various J analysis methods (see Section A2.3.1, Appendix A) with currently available experimental data that describe the moment and J values corresponding to first crack extension (see Table A-3 and figures in Appendix A) for ferritic and stainless steel piping.

- The results from this comparison (see Table A-4 and Figure A-8 in Appendix A) indicate that the Electric Power Research Institute (EPRI) estimation scheme consistently predicted a lower than actual moment at flaw initiation with a maximum difference of about 20 percent for ferritic piping and 30 percent for stainless steel piping. The method described in NUREG/CR-3464 consistently predicted a higher than actual moment at initiation with a maximum difference of about 10 percent for stainless steel and 20 percent for ferritic piping. The NRC modification of NUREG/CR-3464 predicted results that were closer to the actual initiation moment in the majority of cases with a maximum error of about 10 percent overprediction for stainless steel and 20 percent overprediction for ferritic steel.

- Table A-4 and Figure A-7 in Appendix A show that the EPRI estimation scheme consistently overpredicted the value of J at the experimental initiation moment. The computed J values differed by a maximum factor of seven for 16-in.-diameter stainless steel pipe and three for the ferritic pipe. The NUREG/CR-3464 estimation method consistently underpredicted the value of J at initiation. The computed J values differed by a maximum factor of 10 for stainless steel pipe (4-in.-diameter) and 4 for ferritic pipe. The NRC-modified NUREG method underpredicted J in the majority of cases. The computed J values underpredicted by a maximum factor of three for both the stainless steel pipe (4-in.-diameter) and the ferritic pipe.

- The guidelines developed for applying leak-before-break technology (see Section 5.0) are intended to provide adequate margin against full pipe break by selecting reasonably conservative analytical models, material properties, and margins on leak rate, load and flaw size. However, analyses performed as part of this effort indicate that there can be significant differences between experimental results and predictions made by various computational procedures. These differences show that certain computational procedures are sometimes nonconservative; consequently, the analyst must take steps when applying the technology to ensure that nonconservative predictions are not made and the intended overall margins against full pipe break described in this report are maintained.

- When crack extension is predicted to occur, stability analysis should be performed (see Section 5.0) to determine if adequate margins against crack instability are maintained. Stability computations should include crack extension characteristics of the materials as defined by appropriate J-R curve data.

1.0 INTRODUCTION

The purpose of this report is to discuss the subject of nuclear power facility piping integrity in the context of what has become known as "leak-before-break" (LBB) and to make recommendations to the Nuclear Regulatory Commission (NRC) and to the nuclear industry regarding criteria for and the implementation of this general subject. The current NRC regulations as they relate to this subject are discussed and changes to them are suggested to permit the use of advanced fracture mechanics technology (both deterministic and probabilistic) in the licensing process in lieu of requiring the postulation of arbitrary, full double-ended pipe ruptures or their equivalent slot-type pipe breaks. Analyses showing that the likelihood of detectable leaks is significantly greater than a large pipe break demonstrate that the overall safety of a nuclear facility is not jeopardized and indeed can be increased by the elimination of large structural elements previously required for dynamic protection against large breaks. In addition to the significant cost savings, there is a net benefit in terms of reduced man-rem exposure during maintenance and in-service inspection because of the greater accessibility of piping and equipment.

Because leak-before-break technology is relatively new, having matured during the past few years, and is still being refined, this report discusses it in some detail. Guidance is given as to acceptable evaluation models and procedures.

When pipes are demonstrated to have a vanishingly low probability of rupture, it is necessary to address the various Standard Review Plans (SRPs) and Regulatory Guides that may be affected. This report makes recommendations regarding these matters.

If pipes are demonstrated not to break and cause dynamic effects, it can be argued that they will not break for other purposes such as for setting design requirements for containment, emergency core cooling systems (ECCS), maintaining a support structural integrity margin, etc. This report does not consider the latter aspect in order to narrow the application of leak-before-break technology for the time being and hence to expedite its adoption for a limited purpose. This is not to say that a future extension of the technology to address these other aspects is not warranted or is undesirable. It would, however, impact many of the NRC regulations and conceivably could be considered in a rulemaking process in the future.

Following the NRC staff guidance, this report does not address the application of fracture mechanics technology to certain pipes or regions of piping systems that are subject to crack initiation due to thermal fatigue, stress corrosion, or water hammer. The reason for not extending the LBB technology in its entirety to these areas is that its acceptance without other mitigating measures is not compatible with the Commission's defense-in-depth principle. It can be, and is, used in conjunction with other considerations in addressing the stress corrosion cracking phenomenon of BWR piping systems to gain a greater understanding of the problem.[1.1]

REFERENCES

1.1 Pipe Crack Task Group of NRC Piping Review Committee. August 1984. Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants. NUREG-1061, Vol. I.

2.0 CURRENT REGULATORY REQUIREMENTS

The Commission's regulations, as currently implemented by the applicable Standard Review Plans and Regulatory Guides, impose the postulation of piping ruptures in high energy fluid systems, both inside and outside of containment as part of the design bases for structures, systems, and components important to safety. These postulated ruptures include circumferential and longitudinal breaks, up to and including double-ended guillotine breaks in piping which also encompasses the largest pipe in the reactor coolant system. The direct result of such postulated piping ruptures led to the establishment of Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on PWR Primary Systems", and criteria to protect structures, systems, and components important to safety against the consequences of pipe breaks in all other high energy fluid systems. Protective measures include physical isolation from postulated pipe rupture locations if feasible or the installation of pipe whip restraints, jet impingement shields, or compartments.

2.1 EVOLUTION OF REQUIREMENTS

The "design basis accident", "maximum credible accident" or "maximum hypothetical accident" have been used as terms describing what was generally the double-ended guillotine break. The concept was originated by the U.S. Atomic Energy Commission for the multiple purpose of sizing containments and establishing "accident" doses and later, the sizing of emergency core cooling systems. The original concept was quite straightforward; namely, an instantaneous DEGB of a major pipe in the primary system of a light water reactor (LWR) would maximize the fluid release and establish an upper bound for the design pressure established for a containment. This optimized the containment volume vis-a-vis a reasonable design accident pressure.

Later changes in regulatory philosophy, primarily with regard to seismic design, tended to shift the DEGB from a hypothetical accident to one having increasing credence. It was a relatively short step from the hypothetical to a belief in major pipe breaks. A natural consequence of an accepted pipe break was the assumption of a terminal end (reactor pressure vessel nozzle) break and the asymmetric loading of the reactor pressure vessel (Generic Issue A-2). If one accepts a DEGB, then massive pipe restraints to minimize pipe deflection become a natural consequence, and backfitting requirements follow automatically.

A reassessment of the overall probability of a large pipe break, particularly in reactor primary systems, undermined the basic premise that a DEGB was an accepted event. Both probabilistic studies on PWRs (Westinghouse and Combustion Engineering), deterministic studies, and an assessment of failure statistics in large pipes and non-nuclear vessels led to the same conclusion: the probability of a DEGB is extremely low.

A value-impact assessment of backfitting older reactors to incorporate massive pipe restraints indicated a major penalty in man-rem exposure and in installation costs, far out of line with the failure probability and public risk. The current status, based on the preceding studies, does not require backfit on a case-by-case basis for selected plants. This status is subject to further review, in a generic context, by this report.

2.2 REGULATIONS

Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants" to 10 CFR Part 50 requires postulation of pipe breaks and provision of appropriate protection against associated dynamic effects. Note that the regulations also impose other design requirements stemming from postulated pipe breaks, e.g., emergency core cooling system (ECCS) (10 CFR Part 50.46), containment (GDC-16, -50), other engineered safety features (GDC-34, -38, -41) and the environmental qualification of equipment (10 CFR Part 50.49). However, the scope of this report is limited to addressing only the dynamic effects resulting from postulated pipe breaks. In that regard, the effective regulation is GDC-4 in the context of the definition of loss-of-coolant accidents, both of which are reproduced below.

> Criterion 4 - Environmental and missile design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental

conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Loss-of-coolant accidents - Loss-of-coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant make-up system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.*

The footnote to the definition of loss of coolant accidents warrants further discussion. Criteria relating to the type, size and orientation of postulated breaks were developed by the staff, although not promulgated in the regulations. These criteria were published first in a regulatory guide and later in Standard Review Plan (SRP) sections both of which are described below.

2.3 REGULATORY GUIDES

2.3.1 Leak Detection

Early detection of leakage in components of the reactor coolant pressure boundary (RCPB) is necessary to identify deteriorating or failed components and minimize the release of fission products. Consequently, Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", published May 1973, describes acceptable methods to select leakage detection systems for the RCPB. The position of Regulatory Guide 1.45 is that at least three different detection methods should be employed in the reactor. Sump level

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* Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

flow monitoring and airborne particulate radioactivity monitoring are specifically recommended. A third method to be selected may involve either monitoring of condensate flow rate from air coolers or monitoring of airborne gaseous activity. Although these methods used for leak detection still reflect the state of the art, other techniques may be developed and used. Regulatory Guide 1.45 also recommends that flow rates from identified and unidentified sources should be monitored separately, the latter to an accuracy of 1 gpm, and indicators and alarms for leak detection should be provided in the main control room. While leakage limits are not specified, the sensitivity and response time for each leakage detection system used should be capable of detecting 1 gpm or less in one hour.

2.3.2 Pipe Whip

Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment" published in May 1973, constituted the staff's first quantitative basis for selecting the design locations and orientations of postulated breaks in fluid system piping and for determining the measures that should be taken for restraint against pipe whipping. The Guide encompasses both ASME Code Class 1 and 2 piping inside Containment and provides criteria for (1) postulated pipe break locations based on stress or fatigue usage factors as applicable, (2) type of breaks at these locations, (3) measures for restraint against pipe whipping and (4) pressure and temperature conditions in the piping system which constitute high energy levels that could cause whipping if the pipe ruptures.

2.4 TECHNICAL SPECIFICATIONS

General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary" of Appendix A to 10 CFR Part 50 requires in part that, "Means shall be provided for detecting and to the extent practical, identifying the location of the source of reactor coolant leakage." This requirement is implemented via plant technical specifications. The technical specifications limit both unidentified and identified leakage from the reactor coolant system. Since equipment cannot be perfectly leak-tight, allowance is made for identified leakage from valve packing, shaft seals, and other equipment. Thus, even during normal operation, there may be some accumulation of water in the sumps with an increase in the level of radioactivity.

Because the recommendations of Regulatory Guide 1.45 are not mandatory, the technical specifications for 74 operating plants (including BWRs) have been reviewed to determine the type of leak detection methods employed, the range of limiting condition for operation, and the surveillance requirements for the leak detection systems. The American National Standards Institute (ANSI) has prepared a draft standard[2.1] which reviews several leak detection methods and indicates their capabilities for detection, location, and measurement. This information is summarized in Table 2-1. As Table 2-1 indicates, no single currently used leak detection method combines optimal leakage detection sensitivity, leak locating ability, and leakage measurement accuracy.

All plants use at least one of the two systems recommended by Regulatory Guide 1.45. All but eight specify sump monitoring as one of the leakage detection systems, and all but three use particulate radioactivity monitoring. Monitoring drywell air cooler condensate flow rate and atmospheric gaseous radioactivity are also frequently used. Leakage limits for most plants have also been tabulated. The allowed limits on reactor unidentified coolant leakage are shown in Figure 2-1a. The limit for all PWRs is 1 gpm and the limit for most BWRs is 5 gpm. The limits for total leakage (Figure 2-1b) are generally 10 gpm for PWRs and 25 gpm for BWRs. (Regulatory Guide 1.45 does not specify leakage limits, but does suggest that the leakage detection system should be able to detect a 1-gpm leak in 1 hour.) In some cases limits for rates of increase in leakage are stated in the plant technical specifications. On an hourly basis they are either 0.1 gpm/h (2 BWRs) or 0.5 gpm/h (4 BWRs). Additional limits for rates of increase in leakage (2 gpm/24 h) were temporarily imposed on five BWRs as part of the five orders (IGSCC (intergranular stress corrosion cracking) inspection orders confirming shutdown) of August 26, 1983.

Table 2-1 Capabilities of Leakage Monitoring Methods

Method	Leakage Detection Sensitivity	Leakage Measurement Accuracy	Leak Location
Sump Monitoring	G[a]	G	P[c]
Condensate Flow Monitors	G	F[b]	P
Radiogas Activity Monitor	F	F	F
Radioparticulate Activity Monitor	F	F	F
Primary Coolant Inventory[d]	G	G	P
Humidity Dew Point	F	P	P
Tape Moisture Sensors	G	P	G
Temperature	F	P	F
Pressure	F	P	P
Liquid Radiation Monitor[e]	G	F	F
Visual[f]	F	P	G

[a] G (Good) - can generally be applied to meet intent of this standard if properly designed and utilized.

[b] F (Fair) - may be acceptable, marginal, or unable to meet intent of this standard depending upon application conditions and the number of measurement points or locations.

[c] P (Poor) - not normally recommended but might be used to monitor specific confined locations.

[d] For PWR during steady state conditions.

[e] For detection of intersystem leakage; may also be used for location function in sump or drain monitoring.

[f] Provided that the leakage area is visible.

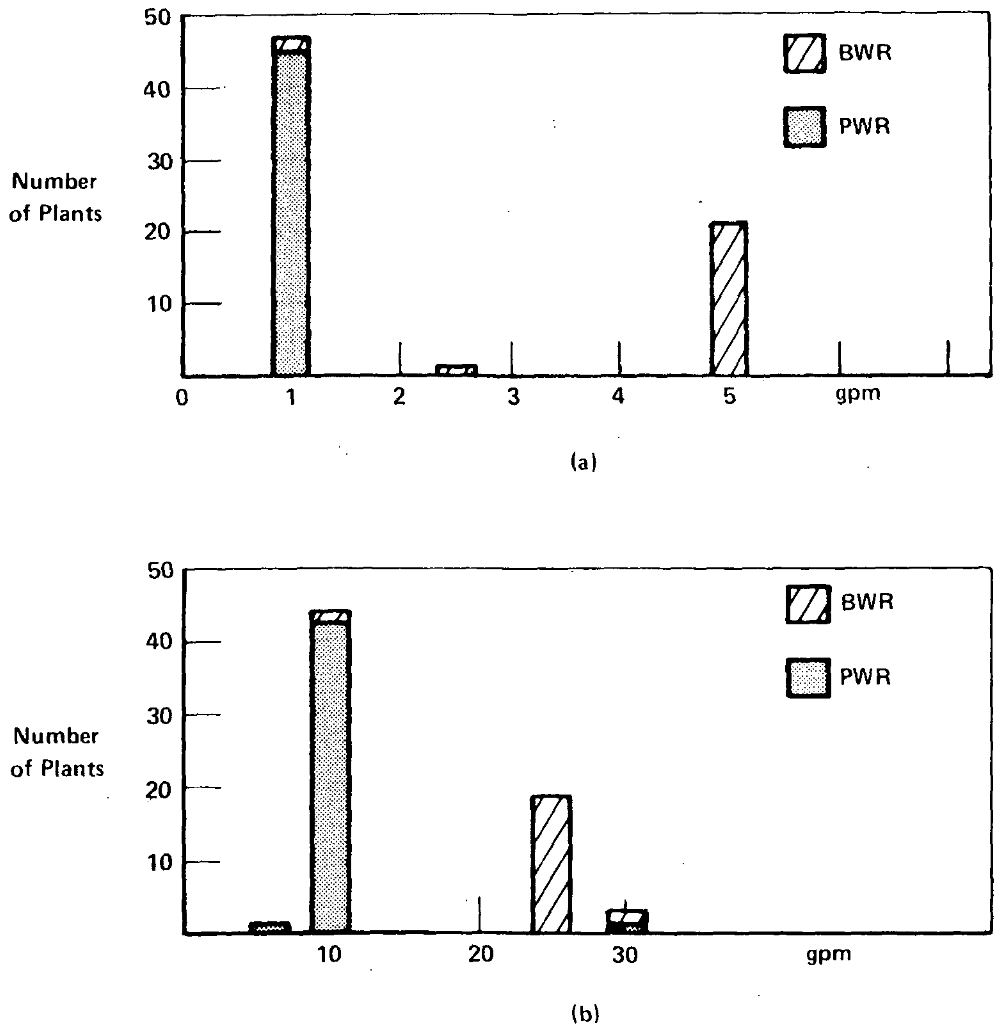
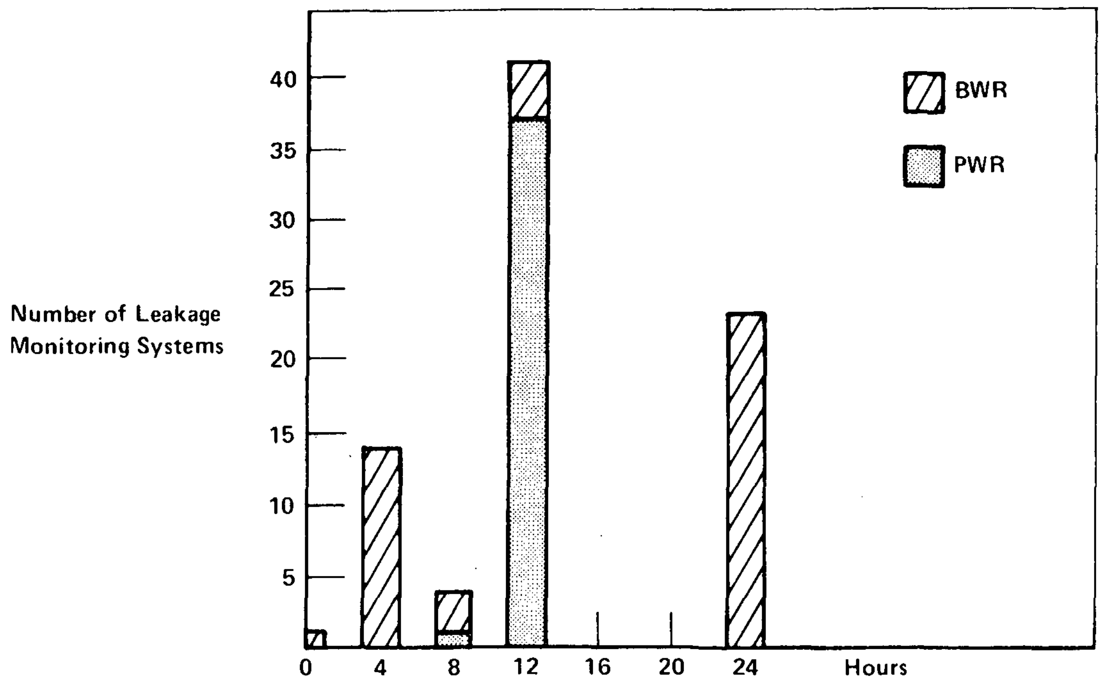
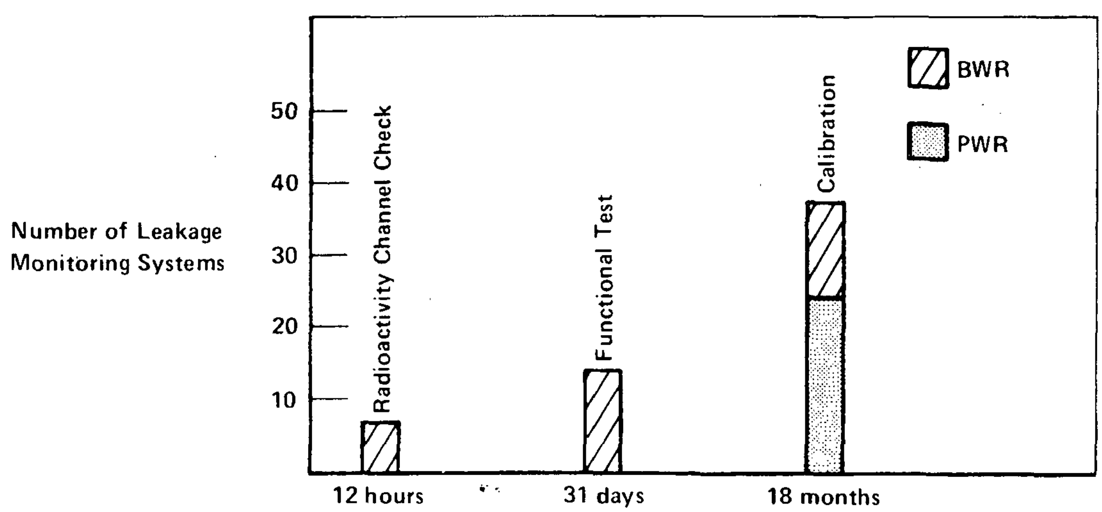


Figure 2-1 Allowed Limits on Reactor Coolant System Leakage from Technical Specifications for 74 Plants for (a) Unidentified Leakage and (b) for Total Leakage



(a)



(b)

Figure 2-2 Histograms Based on the Technical Specifications for 74 Plants and the Number of Leakage Monitor Systems Versus (a) System Check Interval (Hours) and (b) System Calibration or Functional Test Interval

Surveillance periods are indicated in Figure 2-2a. Leakage in most PWRs is checked every 12 hours, and in most BWRs every 4 or 24 hours. One BWR specifies that a continuous monitor with control room alarm shall be operational. The intervals between system calibration and checks are indicated in Figure 2-2b. For BWRs, calibration is generally performed at 18-month intervals and functional tests every month.

In general, sump pump monitoring is used to establish the presence of leaks. Other methods appear to be less reliable or less convenient. In most reactors the surveillance periods are too long to permit detection of a 1-gpm leak in one hour as suggested by Regulatory Guide 1.45, but it appears that this sensitivity could be achieved if monitoring procedures were modified. None of the systems provides any information on leak location, and leaks must be located by visual examination after shutdown. Since cracks may close when the reactor is shut down, reducing flow rates considerably, it would be desirable to be able to locate cracks during plant operation.

The estimated sensitivity of leakage monitoring systems is occasionally addressed in the technical specifications. For example, one specification indicates that air particulate monitoring can, in principle, detect a 0.013-gpm leak in 20 min, that the sensitivity of gas radioactivity is 2 to 10 gpm, and that of condensate flow monitoring is 0.5 to 10 gpm. Sump pump monitoring appears capable of detecting 1-gpm leak in 10 to 60 minutes (with continuous monitoring).

The impact of reactor coolant pressure boundary (RCPB) leakage detection systems on safety was evaluated for eight reactors as part of the Integrated Plant Safety Assessment-Systematic Evaluation Program (NUREG 0820-0827). In four of the eight reactors a 1-gpm leak would not be detected in 1 hour nor did they have three leakage monitoring systems, as suggested by Regulatory Guide 1.45. The fracture mechanics and leak rate calculations in the Systematic Evaluation Program (SEP) plants indicate that current leak detection systems and leakage limits will detect and require plant action for throughwall cracks 4 to 10 in. long in 12- to 28-in.-diameter piping in one day. Since these cracks are much smaller than those required to produce failure in tough reactor piping, improved leak detection systems may offer little safety benefit for this particular class of flaws when crack growth occurs by a relatively slow mechanism. Although current leak detection systems are adequate to ensure leak before break in a great majority of cases, local leak detection systems may be desirable for some postulated break locations where separation and/or restraint is not practical to remove the effects of a high energy pipe break.

There are some shortcomings in existing leak detection systems. The Duane Arnold safe end cracking incidents indicate that the sensitivity and reliability of current leak detection systems are clearly inadequate in some cases. The plant was shut down on the judgment of the operator when a leak rate of 3 gpm was detected; however, this rate is below the required shutdown limit for almost all BWRs. Examination of the leaking safe end showed that cracking had occurred essentially completely around the circumference. The crack was throughwall about 20 percent of the circumference and 50 to 75 percent throughwall in the nonleaking areas. The other seven riser safe ends were also severely cracked, but since the cracks were not throughwall no leakage resulted.

Simply tightening the current leakage limits may not be an adequate solution to these shortcomings, since it is possible that this may produce an unacceptably high number of spurious shutdowns due to the inability of current leak detection systems to identify leak sources.

One other safety-related aspect of improved leak detection systems is in the area of radiation exposure to plant personnel. Improved systems with leak location capability could reduce the exposure of personnel inside containment. Some welds are inaccessible for inspection and improved leak detection would provide additional margin in terms of early detection of leakage. Improved leak detection is consistent with the defense-in-depth philsophy of the NRC and would lead to earlier detection of system degradation.

Note that there are no requirements for, nor do the technical specifications cover, leakage detection for systems other than the reactor coolant system.

2.5 STANDARD REVIEW PLAN (SRP) Sections

SRP Sections 3.6.1, 3.6.2 and 3.9.3 will be revised to incorporate current staff positions.

2.6 GENERIC ISSUE (A-2)

The problem of asymmetric blowdown loads on PWR primary systems, initially identified to the staff in 1975, was designated Unresolved Safety Issue (USI) A-2 and is described in detail in NUREG-0609 which provides a pressure-load analysis method acceptable to the staff. This issue deals with safety concerns following a postulated major double-ended pipe break in the primary system. Previously unanalyzed loads on primary system components have the potential to alter primary system configurations or damage core cooling equipment and contribute to core melt accidents. For postulated pipe breaks in the cold leg, asymmetric pressure changes could take place in the annulus between the core barrel and the reactor pressure vessel (RPV). Decompression could take place on the side of the reactor pressure vessel annulus nearest the pipe break before the pressure on the opposite side of the RPV changed. This momentary differential pressure across the core barrel induces lateral loads both on the core barrel itself and on the reactor vessel. Vertical loads are also applied to the core internals and to the vessel because of the vertical flow resistance through the core and asymmetric axial decompression of the vessel. For postulated, essentially instantaneous breaks in the RPV nozzles, the annulus between the reactor and biological shield wall could become asymmetrically pressurized, resulting in additional horizontal and vertical external loads on the reactor vessel. In addition, the reactor vessel is loaded simultaneously by the effects of strain energy release and blowdown thrust at the pipe break. For similar breaks at reactor vessel outlet nozzles, the same type of loadings could occur, but the internal loads would be predominantly vertical because of the more rapid decompression of the upper plenum. Similar asymmetric forces could also be generated by postulated pipe breaks located at the steam generator and reactor coolant pump.

The resolution of this issue would have required some licenses for operating PWRs to add massive piping restraints to prevent postulated large pipe ruptures from resulting in full double-ended pipe break, thus reducing the blowdown asymmetric pressure loads and the need to modify equipment supports to withstand those loads as determined in plant-specific analysis (e.g., WCAP-9628 and WCAP-9748, "Westinghouse Owners Group Asymmetric LOCA

Loads Evaluation"). Instead, this issue was resolved by the industry and the NRC staff by adoption of the leak-before-break approach utilizing advanced fracture mechanics techniques as discussed in the following sections of this report.

REFERENCES

2.1 The American National Standards Institute. 1978. Standard for Light Water Power Reactor Coolant Pressure Boundary Leak Detection. ISA Standard S67.03, ANS Standard N41.21.

3.0 CURRENT AND ONGOING STAFF ACTIONS

3.1 EVOLUTION OF FRACTURE MECHANICS TECHNOLOGY

Subsequent to identification in 1975 of the generic safety concern (i.e., loads from postulated pipe ruptures in PWR reactor coolant main loop piping) that initiated Unresolved Safety Issue (USI) A-2, the fracture mechanics technology regarding the potential rupture of tough piping such as used in PWR primary coolant systems, has advanced considerably. The behavior of piping with flaws, either postulated or real, under normal and accident loads is now better understood.

Also in the interim, in recognition of the various negative impacts on plant design and in-service inspection, the NRC, via its contractors, and the industry have spent significant time and effort to develop advanced fracture mechanics technologies applicable to pressure-retaining components including piping systems. These technologies are based on theory and validation by experiments. The conclusion reached from these studies and from many reactor years of operating experience is that flawed piping is much more likely to leak before it breaks.

These advanced fracture mechanics techniques deal with relatively small flaws in piping components (either postulated or real) and examine their behavior under various pipe loads. The objective is to demonstrate by deterministic analyses that the detection of small flaws either by in-service inspection or by leakage monitoring systems is assured long before the flaws can grow to critical or unstable sizes which could lead to large break areas such as the double-ended loss-of-coolant accident (LOCA) or its equivalent. The concept underlying such analyses is referred to as "leak before break". There is no implication that piping failures cannot occur, but rather that improved knowledge of the failure modes, of piping systems and the application of appropriate remedial measures if indicated, can reduce the probability of catastrophic failure to insignificant values.

3.2 USI A-2 RESOLUTION

3.2.1 Topical Report Evaluation

Advanced fracture mechanics technology was applied in proprietary topical reports WCAP 9558, Rev. 2, and WCAP 9787, both dated May 1982, which were submitted to the NRC staff by Westinghouse on behalf of 11 licensees (16 operating units) belonging to the A-2 Owners Group. The topical reports for those licensees' plants were intended to resolve the issue of asymmetric blowdown loads on the PWR primary systems that resulted from a limited number of discrete break locations as stipulated in the resolution of USI A-2. However, the topical reports also demonstrated that main loop primary coolant piping breaks would not occur at any location, thus eliminating any possible need for installation of pipe whip restraints or jet impingement shields.

In its evaluation of the Westinghouse topical reports, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack", WCAP 9558, Rev. 2, and "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation", WCAP 9787, the NRC staff concluded that large margins against unstable crack extension exist for certain stainless steel PWR primary coolant piping postulated to have large flaws and subject to the safe shutdown earthquake (SSE) in combination with the loads associated with normal plant conditions.

3.2.2 Action by the NRC Committee to Review Generic Requirements (CRGR)

Because the application of leak-before-break technology in lieu of postulated large pipe ruptures is at variance with current NRC regulations, the proposed staff actions regarding USI A-2 were presented to CRGR. The NRC staff developed a package for CRGR review which included (a) the staff's topical report evaluation containing technical justification for granting exemptions from GDC-4, and (b) the regulatory (value-impact) analysis. (Both these documents are attached as Enclosures 1 and 2 respectively to NRC Generic Letter 84-04 dated February 1, 1984, which constitutes the NRC position regarding the Westinghouse Owner's Group facilities.)

3.2.3 Regulatory Analysis

As part of the resolution of USI A-2 a regulatory analysis was performed which supported the staff's proposed exemptions to the regulations for CRGR review. This analysis is discussed in detail in Section 6.0, "Value-Impact". It concludes that the savings, both in terms of occupational radiation exposure and costs far outweigh any potential benefits (e.g., decrease in public risk and avoided accident exposure) from plant modifications.

3.2.4 CRGR Recommendations

The NRC staff met with the CRGR to review the issue on September 28, 1983. In the minutes of that meeting dated October 14, 1983, the CRGR recommended that the Executive Director for Operations (EDO) accept the staff's technical findings and proposed actions. The CRGR observed that these findings and the technical justifications in support of the findings could extend to other break locations and to assumptions previously made for piping loops and components of the reactor coolant system, for piping connected to the coolant system, and perhaps to the piping of other systems in the plant. To maximize the utility of the staff's recommendation and their potentially positive benefits to plants under construction, the CRGR recommended a special staff effort to implement these recommendations to the extent justifiable in terms of safety and staff resources. The preceding was summarized from "Minutes of CRGR Meeting Number 47", dated October 14, 1983, memorandum from V. Stello, Jr., to W. J. Dircks.

3.2.5 Exemptions

As a result of its review, the CRGR in its recommendations to the EDO endorsed the staff's position that an acceptable technical and regulatory basis exists to grant exemptions to General Design Criterion 4 (GDC-4) in regard to providing protection against asymmetric blowdown loads. The scope and bases for these exemptions were specified in Generic Letter 84-04 issued February 1, 1984, to all PWR licensees, construction permit holders, and applicants for construction permits. The subject of the generic letter was "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops".

Scope and Bases. Generic Letter 84-04 provides the scope and bases for exemptions to GDC-4 as follows:

> Authorization by NRC to remove or not to install protection against asymmetric dynamic loads (e.g., certain pipe whip restraints) in the primary main coolant loop will require an exemption from GDC-4. Licensees must justify such exemptions on a plant-by-plant basis. In such exemption requests, licensees should perform a safety balance in terms of accident risk avoidance attributable to protection from asymmetric blowdown loads versus the safety gains resulting from a decision not to use such protection. In the latter category are (1) the avoidance of occupational exposures associated with use of and subsequent removal and replacement of pipe whip restraints for in-service inspections, and (2) avoidance of risks associated with improper reinstallation. Provided such a balance shows a net safety gain for a particular facility, an exemption to GDC-4 may be granted to allow removal of existing restraints or non-installation of restraints which would have otherwise been required to accommodate double-ended break asymmetric dynamic loading in the primary coolant loop.
>
> Other PWR licensees or applicants may also request exemptions on the same basis from the requirements of GDC-4 with respect to asymmetric blowdown loads resulting from discrete breaks in the primary main coolant loop, if they can demonstrate the applicability of the modeling and conclusions contained in the referenced reports to their plants or can provide an equivalent fracture mechanics-based demonstration of the integrity of the primary main coolant loop in their facilities.

3.3 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) ENDORSEMENT

After hearing a report from its Subcommittee on Metal Components and presentations from the staff and its consultants (Lawrence Livermore Laboratory), the ACRS acknowledged the staff's leadership in validating the application of fracture mechanics to nuclear systems and components. The ACRS further stated, "Over the last decade this [fracture mechanics] has led to a sound basis for predicting the conditions under which cracks in the primary pressure boundary will be stable. In particular, this work has provided confidence in predicting the range of crack sizes that will be stable and grow slowly. That is, crack sizes that will leak but not break."(3.1)

3.4 LIVERMORE PROBABILISTIC DEGB PROGRAM

3.4.1 Purpose and Scope

The Lawrence Livermore National Laboratory (LLNL), through its Nuclear Systems Safety Program, is performing probabilistic reliability analyses of PWR and BWR reactor coolant piping for the NRC Office of Nuclear Regulatory Research. Specifically, LLNL is estimating the probability of a double-ended guillotine break (DEGB) in the reactor coolant loop piping in PWR plants, and in the main steam, feedwater, and recirculation piping of BWR plants. For these piping systems, the results of the LLNL investigations will provide NRC with a technical basis with which to

(1) Reevaluate the current general design requirement that DEGB be assumed in the design of nuclear power plant structures, systems, and components.

(2) Determine if an earthquake could induce a DEGB, and thus reevaluate the current design requirement that pipe break loads be combined with loads resulting from a safe shutdown earthquake (SSE).

(3) Make licensing decisions concerning the replacement, upgrading, or redesign of piping systems, or addressing such issues as the need for pipe whip restraints on reactor coolant piping.

In estimating the probability of DEGB, LLNL considers two causes of pipe break: pipe fracture due to the growth of cracks at welded joints ("direct" DEGB), and pipe rupture indirectly caused by the seismically induced failure of critical supports or equipment ("indirect" DEGB).

Although these investigations are limited to the reactor coolant piping noted above, the techniques used to assess reliability are sufficiently general that they could be conveniently applied to other piping systems not included in the present LLNL investigations.

3.4.2 Status

To arrive at a general conclusion about the probability of DEGB in the reactor coolant loop piping of PWR plants, LLNL is taking a vendor-by-vendor approach. For each of the three PWR vendors (Westinghouse, Babcock & Wilcox, and Combustion Engineering) the principal tasks are to

(1) Estimate the probability of direct DEGB taking into account such contributing factors as initial crack size, pipe stresses due to normal operation and sudden extreme loads (such as earthquakes), the crack growth characteristics of pipe materials, and the capability to nondestructively detect cracks, or to detect a leak if a crack penetrates the pipe wall. To do this LLNL developed a Monte Carlo simulation methodology, implemented in the PRAISE computer code (see Appendix B).

(2) Estimate the probability of indirect DEGB by identifying critical component supports or equipment whose failure could result in pipe break, determining the seismic "fragility" (relationship between seismic response and probability of failure) of each, and combining this result with the probability that an earthquake occurs exceeding a given level of ground acceleration ("seismic hazard").

(3) For both causes of DEGB, perform sensitivity studies to identify key parameters contributing to the probability of pipe break.

(4) For both causes of DEGB, perform uncertainty studies to determine how uncertainties in input data affect the uncertainty in the final estimated probability of pipe break.

LLNL has completed generic evaluations of DEGB probability for plants with nuclear steam supply systems manufactured by Westinghouse and by Combustion Engineering.[3.2 - 3.4] The results of these evaluations indicate that the probability of DEGB from either cause is very low. Therefore, this result suggests that the DEGB design requirement -- and with it related design issues such as coupling of DEGB and SSE loads, asymmetric blowdown, and the need to install pipe whip restraints -- warrants a reevaluation for PWR reactor coolant loop piping.

In the Westinghouse and Combustion Engineering evaluations, LLNL designated a single reference, or "pilot" plant, as a basis for methodology development as well as for extensive sensitivity studies to identify the influence that individual parameters have on DEGB probabilities. Thus, each pilot plant was used to develop and "shake down" the assessment methodology that was later applied in the corresponding generic study for each vendor.

In the generic study of reactor coolant piping manufactured by each of these vendors, LLNL evaluated individual plants, or groups of plants sharing certain common or similar characteristics, to arrive at an estimated DEGB probability (including uncertainty bounds) characteristic of all plants. Thus, the generic evaluation represented a "production" application of the assessment methodology.

The objectives and approach of the BWR study are essentially the same. LLNL is currently limiting its investigation to Mark I plants, which have recirculation piping particularly susceptible to the effects of intergranular stress corrosion cracking (IGSCC), and is beginning with a pilot study based on the Brunswick plant operated by Carolina Power & Light. As part of the BWR investigation, LLNL is developing a probabilistic IGSCC model which will consider crack initiation as well as the effect of stress corrosion on pre-existing cracks; a prototype has been completed and implemented in the PRAISE code. LLNL is also developing a PRAISE model to consider stress redistribution among weld joints due to the failure of intermediate pipe supports; this was unnecessary in the PWR evaluations because reactor coolant loop piping is supported solely by the loop components. The BWR pilot study is scheduled for completion by October 1984.

For Babcock & Wilcox PWR plants, LLNL is estimating the probability of indirect DEGB for each of two representative plants: one plant with the raised loop nuclear steam supply system, and one plant with the lowered loop configuration. The probability of indirect DEGB will not be evaluated for every B&W plant; instead, LLNL will collect information on component support strength for all plants, and will perform sensitivity studies which will provide insight on the degree to which the results of the plant-specific indirect DEGB evaluations are characteristic for all B&W plants. LLNL is also obtaining and reviewing information required for an evaluation of direct DEGB for the representative raised loop plant.

3.4.3 Results

Probability of Direct DEGB in Reactor Coolant Loop Piping. LLNL has completed probabilistic analyses indicating that the probability of direct DEGB in reactor coolant piping is very small for Westinghouse PWR plants located east of the Rocky Mountains (see Table 3-1). These analyses calculated the growth of as-fabricated surface flaws at welded joints, taking into account loads on the piping due to normal operating conditions and seismic events. Other factors, such as the capability to detect cracks by nondestructive examination and the capability to detect pipe leaks, were also considered.

In particular, the results of the evaluations for 17 sample plants (33 actual plants) indicate that

- The "best estimate" probability of direct DEGB ranges from 1×10^{-12} to 6×10^{-12} events per plant-year, with a median value (50 percent confidence limit) of about 4.4×10^{-12} events per plant-year.
- The "best estimate" probability of a 3-gpm leak (throughwall crack) ranges from 5×10^{-9} to 5×10^{-7} events per plant-year, with a median value of about 1.1×10^{-7} events per plant-year. Actually, there have been no reported leaks in these large PWR PCS pipes. The significantly greater probability of leak compared to DEGB supports the concept of leak before break in PWR reactor coolant loop piping.

Table 3-1 Annual Probabilities of Direct DEGB and Leak for Westinghouse PWR Plants (events per plant-year)

	Lower Bound[a]	Median[b]	Upper Bound[c]
All Eastern and Midwestern Plants[d]			
DEGB	5.0×10^{-17}	4.4×10^{-12}	7.5×10^{-10}
Leak	5.6×10^{-10}	1.1×10^{-7}	2.4×10^{-7}
West Coast Plants[e]	Evaluation in progress		

(a) Sample plant with lowest probability of DEGB, 10 percent statistical confidence limit.

(b) Median best-estimate value for all sample plants considered.

(c) Sample plant with highest DEGB probability, 90 percent statistical confidence limit.

(d) Generic seismic hazard curves used in evaluation.

(e) Site-specific seismic hazard curves used in evaluation

- Uncertainty analyses indicated that the upper bound values of DEGB and leak probabilities are 1.2×10^{-10} and 2×10^{-7} events per plant-year, respectively.

Similar analyses are nearing completion for Westinghouse plants located on the west coast. Preliminary results indicate best-estimate break probabilities on the order of 10^{-9} events per reactor-year.

The results of the LLNL generic study of Combustion Engineering PWR plants indicated that the probability of a direct DEGB in reactor coolant loop piping is equally low (see Table 3-2). An interesting result was that the probability of direct DEGB for the carbon steel piping used in these plants was typically higher than that for the more ductile stainless steel piping used in the Westinghouse plants, if the effects of nondestructive examination were neglected. However, the greater certainty of crack detection in carbon steel roughly equalizes the direct DEGB probabilities for the two types of reactor coolant loop systems, a clear illustration of the ability of probabilistic techniques to consider how the interaction of seemingly unrelated parameters can affect overall pipe reliability.

The results of this study also indicated that the probability of an earthquake causing a direct DEGB is as negligible for Combustion Engineering reactor coolant loop piping as it is for the eastern Westinghouse plants.

Probability of Indirect DEGB in Reactor Coolant Loop Piping. LLNL has completed probabilistic analyses for 46 Westinghouse plants located east of the Rocky Mountains indicating that the probability of indirect DEGB in reactor coolant loop piping is very small for these plants (see Table 3-3). In evaluating the probability of indirect DEGB for each plant, critical components were first identified and the seismic "fragility" of each estimated. For each component, the probability that its failure could lead to DEGB was then determined. Finally, the non-conditional probability of indirect DEGB was estimated by statistically combining generic seismic hazard curves for the eastern U.S. with a "plant level" fragility derived from the individual component fragilities.

Table 3-2 Annual Probabilities of Direct DEGB and Leak for Combustion Engineering PWR Plants (events per plant-year)

	Leak[a]	DEGB[a]
Palo Verde 1,2,3	1.5×10^{-8}	4.5×10^{-13}
San Onofre 2,3	2.2×10^{-8}	1.0×10^{-13}
WPPSS 3	1.8×10^{-8}	6.1×10^{-14}
Waterford	1.8×10^{-8}	9.0×10^{-14}
Group "A" Composite[b]	2.3×10^{-8}	5.5×10^{-14}

(a) Best-estimate probabilities.

(b) Group "A" is a Combustion Engineering designation for the following facilities:

Calvert Cliffs 1, 2
Millstone 2
Palisades
St. Lucie 1, 2.

The results of these analyses indicated for Westinghouse plants east of the Rocky Mountains that

- The critical components whose failure would result in DEGB were the reactor pressure vessel supports, the reactor coolant pump supports, and the steam generator supports. For the Zion Unit 1 plant used in the pilot study, the overhead crane in the containment building was also a critical component due to its atypical design. More typical crane designs, supported near the containment dome, did not contribute significantly to the probability of indirect DEGB.

Table 3-3 Annual Probabilities for Indirect DEGB for Westinghouse PWR Plants (Events Per Plant-Year)

	Confidence Limit[a] 10%	50%	90%
Lowest Seismic Capacity Eastern Plants[b]			
Designed for SSE[c] + DEGB[d]	2.3×10^{-7}	3.3×10^{-6}	2.3×10^{-5}
Designed for SSE alone	1.0×10^{-7}	2.4×10^{-6}	2.0×10^{-5}
All 46 Eastern Plants[b]	2.0×10^{-9}	1.0×10^{-7}	7.0×10^{-6}
West Coast Plants[e]			
San Onofre Unit 1			
SONGS Set 1	3×10^{-10}	5×10^{-8}	1×10^{-6}
SONGS Set 2	1×10^{-7}	4×10^{-6}	5×10^{-5}
Diablo Canyon Units 1,2	4×10^{-7}	1.7×10^{-6}	2×10^{-5}
Median for West Coast Plants	2×10^{-7}	3×10^{-6}	5×10^{-5}

(a) A confidence limit of 90 percent implies that there is a 90 percent subjective probability (confidence) that the probability of indirect DEGB is less than the value indicated.

(b) Generic seismic hazard curves used in evaluation.

(c) SSE--Safe Shutdown Earthquake

(d) DEGB--Double-ended guillotine break.

(e) Site-specific seismic hazard curves used in evaluation.

- The best-estimate probability of indirect DEGB (50 percent confidence limit) is about 10^{-7} events per plant-year, with an upper bound (90 percent confidence limit) of 7.0×10^{-6} events per plant-year.

- The best-estimate probability of indirect DEGB for one "lower bound" plant designed for the combination of safe shutdown earthquake (SSE) and DEGB loads was 3.3×10^{-6} events per plant-year, with an upper bound (90 percent confidence limit) of 2.3×10^{-5} events per plant-year.

- The best-estimate probability of indirect DEGB for another lower bound plant designed for SSE alone (no DEGB loads) was 2.4×10^{-6} events per plant-year, with an upper bound of 2.0×10^{-5} events per plant-year.

- Only gross design and construction errors of implausible magnitude could substantially increase the probability of indirect DEGB beyond the values predicted.

LLNL also estimated the probabilities of DEGB for two west coast Westinghouse sites -- San Onofre Unit 1 and Diablo Canyon -- using site-specific seismic hazard curves developed from the results of several independent seismic hazard evaluations. The results of these analyses indicated that

- The median probability of indirect DEGB in the reactor coolant piping of west coast plants is about 3×10^{-6} events per plant-year, with an upper bound of about 5×10^{-5} events per plant-year. These values are slightly more than one order of magnitude higher than the corresponding generic probabilities for the plants east of the Rocky Mountains.

- As part of the San Onofre evaluation, LLNL applied two sets of seismic hazard curves. The first ("SONGS set 1") was a best-estimate curve which showed that maximum peak ground acceleration asymptotically approached 1.5 times the SSE.[3.5] Because this best-estimate curve did not include larger earthquakes, LLNL performed a sensitivity evaluation in which this curve was extrapolated to include earthquakes up to five times the SSE ("SONGS set 2"). The median indirect DEGB probabilities estimated using the second set of curves increased by about two orders of magnitude -- from 5×10^{-8} to 4×10^{-6} events per plant-year -- over those predicted using the first set. This result indicates, not surprisingly, that the probability of indirect DEGB is a strong function of seismic hazard. This contrasts with the results of the direct DEGB evaluations, which showed that the break probability is only weakly affected by earthquakes.

The probability of DEGB indirectly caused by the seismically induced failure of heavy component supports is about five orders of magnitude greater than DEGB due to crack growth at welded joints. Thus, the LLNL analyses clearly point to indirect causes as the dominant mechanism leading to DEGB in reactor coolant loop piping.

An evaluation of Combustion Engineering plants indicated the same general results, with the probabilities of indirect DEGB in reactor coolant loop piping typically lower than for the Westinghouse plants (Table 3-4).

3.4.4 Conclusions

Effect of Earthquakes on DEGB Probabilities. The LLNL investigations have shown that the probability of direct DEGB is only very weakly affected by an earthquake. In evaluating the probability of direct DEGB, three events were considered in which failure occurs in reactor coolant loop piping:

- Failure occurs with no earthquake occurring during plant life.
- Failure occurs prior to the first earthquake occurring during plant life.
- Failure occurs simultaneously with the first earthquake occurring during plant life.

Cumulative probabilities of direct DEGB were calculated independently for each event and then combined into an overall cumulative probability that pipe failure occurs sometime during plant life. It was found for both leak and DEGB that the probability of the third event -- simultaneous occurrence of failure and an earthquake -- was three to four orders of magnitude less than that of failure occurring independently of an earthquake. This result indicates that direct DEGB and a safe shutdown earthquake can be considered independent random events, and that the probability of their simultaneous occurrence during plant life is negligibly low -- about 2.1×10^{-12} events for the sample plant with the highest DEGB probability.

Table 3-4 Annual Probabilities of Indirect DEGB Combustion Engineering PWR Plants (Events Per Plant-Year)

	Confidence Limit[a]		
	10%	50%	90%
Palo Verde 1,2,3[b],[c]			
Site-Specific	4.0×10^{-19}	3.8×10^{-16}	1.0×10^{-13}
Generic	2.4×10^{-12}	5.4×10^{-10}	1.1×10^{-7}
San Onofre 2,3[c]			
Site-Specific Set 1	3.5×10^{-18}	4.6×10^{-17}	3.2×10^{-14}
Site-Specific Set 2	5.0×10^{-17}	1.1×10^{-11}	2.1×10^{-9}
WPPSS 3[b]	8.0×10^{-11}	2.9×10^{-9}	1.5×10^{-7}
Waterford[b]	1.1×10^{-10}	1.3×10^{-8}	3.0×10^{-7}
Group "A" Plants[d]	9.0×10^{-10} to 5.0×10^{-7}	6.6×10^{-8} to 1.4×10^{-6}	1.2×10^{-6} to 1.1×10^{-5}

(a) A confidence limit of 90 percent implies that there is a 90 percent subjective probability (confidence) that the probability of indirect DEGB is less than the value indicated.

(b) Generic seismic hazard curves used in evaluation.

(c) Site-specific seismic hazard curves used in evaluation

(d) Refer to Note (b), Table 3-2.

Sensitivity analyses performed during the Zion pilot study for the joint between the hot leg and the RPV indicated less than one order of magnitude difference between DEGB probability when no earthquake was considered and that predicted assuming an earthquake with a peak ground acceleration five times the 0.17g SSE. This result implies that the probability of direct DEGB in reactor coolant piping is only a weak function of earthquake intensity.

The Zion pilot study identified earthquakes as the only credible cause of indirect DEGB; the probability of indirect DEGB therefore also expresses the probability that DEGB and an earthquake simultaneously occur.[3.6] For the lowest capacity Westinghouse plant east of the Rocky Mountains, the upper bound probability (90-percent confidence limit) is 2.3×10^{-5} events per plant-year. The upper bound probability generically applicable to all plants in this region is about 7×10^{-6} events per plant-year, compared to an upper bound value of 5×10^{-5} for west coast plants. Not surprisingly, the sensitivity studies described in Section 3.4.3 indicated that seismic hazard has a significant effect on the estimated probability of indirect DEGB.

In developing the indirect DEGB results, LLNL conservatively assumed that failure of any critical support unconditionally led to DEGB. In other words, no credit was taken for large inelastic deformation of the pipe that might occur resulting in only partial break or no break at all. Furthermore, the wide spread of uncertainty in the generic seismic hazard curves, combined with the assumption of a 0.15g minimum SSE, is expected to cover all sites in the eastern and midwestern U.S. Evaluations of specific plants in areas of low seismicity, using site-specific hazard information, would result in lower plant-specific DEGB probabilities.

<u>Reliability of Heavy Component Supports</u>. If the probability of DEGB is determined to be acceptably low, then the current regulatory requirement that SSE and pipe rupture loads be combined in the design of reactor coolant loop piping could be eliminated. Given that future reactors may not be designed for this load combination, a question may arise concerning the reliability of heavy component supports.

Interestingly, the results of the indirect DEGB evaluation imply that the reliability of heavy component supports is as much a function of the particular analysis techniques used in plant design as it is of load combination. In the study of eastern and midwestern plants, LLNL selected two "lower bound" (lowest seismic capacity) plants for detailed evaluation of component seismic fragilities. For one of these plants, an older plant not designed for the SSE and DEGB load combination, LLNL actually predicted a slightly lower best-estimate probability of DEGB than for the more modern plant that had been designed for both SSE and DEGB loads (2.4×10^{-6} compared to 3.3×10^{-6} events per plant-year, respectively). The older plant had high seismic margins because of relatively conservative analytical techniques used in its design (three-dimensional uncoupled response spectrum analysis). The newer plant, on the other hand, was designed using more sophisticated analytical techniques (three-dimensional coupled time-history response analysis). Although this plant was designed for combined SSE and DEGB loads, reduced conservatism in the analysis methods used yielded a DEGB probability similar to that of the older plant.

The lesser degree of refinement in the design methods for the older plant is, not surprisingly, evidenced by the somewhat larger uncertainty in its DEGB probability.

It can be argued that eliminating the requirement to combine SSE and DEGB loads in the design of component supports will result in less conservative support designs. Load definition is certainly one way of introducing conservatism into an analysis. However, many other factors also contribute to the degree of conservatism in a component design including

- The particular analytical techniques used to predict component response, such as two- or three-dimensional analysis, time-history or response spectrum analysis, coupled or uncoupled analysis, and the various combinations thereof.
- Input data, that is, selection of parameters such as damping values.
- Application of safety factors to calculated results to "ensure" conservatism.

Just what constitutes a "conservative" analysis is therefore subject to debate. For example, best-estimate calculations, using state- of-the-art modeling and realistic response characteristics (damping, for example) can be performed to determine response to conservative design-basis loads. On the other hand, less sophisticated analysis techniques can be used, and conservatism introduced through the input parameters (again, such as damping) that are selected. The example previously discussed illustrates a case where two different approaches to component design yield predicted reliabilities that are remarkably similar.

From this comparison it can be concluded that component support reliability should not be judged solely on the basis of whether or not SSE and DEGB loads are combined. Instead, support reliability should be evaluated in terms of adequate margin against failure, with the definition of "adequate" taking into consideration a wide range of parameters as was done in developing component fragilities for the LLNL indirect DEGB evaluation (see Table 3-5). As was discussed earlier, probabilistic analysis techniques are particularly well-suited for this purpose.

Combination of Seismic and LOCA Effects. How we postulate a pipe break potentially affects how the following aspects of plant design are treated:

- Whipping of broken pipe ends.
- Coolant discharge rate, which in turn sets the minimum make-up capacity of emergency core cooling systems.
- External loads on the reactor vessel and loads on RPV internals resulting from decompression waves.
- Jet impingement loads on structures and equipment in the immediate break vicinity.
- Reaction loads at support locations.
- Global environmental effects -- pressure, temperature, humidity-- affecting containment design as well as the performance of mechanical and electrical equipment important to safety.
- Local environmental effects affecting equipment performance.

Because a loss-of-coolant accident could have long-term as well as short-term effects, it may not necessarily be possible to decouple all seismic and LOCA effects even though the events themselves may not occur simultaneously. For example, in its specifications for environmental qualification of mechanical

Table 3-5 Parameters Considered in Developing Component Fragilities

Structural Response

- Ground spectrum used for design
- Structural damping
- Site characteristics (rock or soil, shear wave velocity, thicknesses of different strata)
- Fundamental frequency of internal structure if uncoupled analysis was performed
- Interface spectra for NSSS[a] points of connection to structure if uncoupled analysis was conducted
- Input ground spectra resulting from synthetic time history applied to structural model

NSSS Response

- Method of analysis (time history or response spectrum, etc.)
- Modeling of NSSS and structure (coupled or uncoupled)
- NSSS system damping
- NSSS fundamental frequency or frequency range
- If uncoupled analysis was performed, whether envelope or multi-support spectra were used.

(a) NSSS--Nuclear steam supply system.

and electrical equipment, Kraftwerk Union (KWU) divides a LOCA in containment into three time regimes:

- A short-term regime (0 to 3 hours after break), in which peak pressure and temperature are reached approximately 10 sec after break, affecting structures as well as those components that would be required either at the time of or immediately following a pipe break.
- An intermediate-term regime (3 to 24 hours after break), which addresses equipment that would be required during the initial recovery phase following a LOCA.

- A long-term regime (over 24 hours after break), addressing in particular corrosion effects on components either required indefinitely or that would be restarted after extended shutdown for later plant reactivation. The maximum period of interest is defined on a component-specific basis, but is generally on the order of several months to a year.

The short-term regime includes the most dynamic effects associated with a LOCA -- pipe whip, jet impingement, decompression waves -- which would result in the most severe LOCA loads. If DEGB were eliminated as a design basis event, then pipe whip could be similarly eliminated, as without a double-ended break the pipe would retain geometric integrity.

Experimental research, in particular full-scale blowdown testing at the HDR facility in West Germany, has shown that loads due to jet impingement and decompression waves in effect coincide with the blowdown event.[3.7] Therefore, if DEGB and earthquake can be considered as independent random events, loads associated with jet impingement and decompression waves could likewise be decoupled from seismic loads.

This may not be the case, however, for other LOCA effects acting over longer or later time periods. Testing at HDR has shown that containment pressure and temperature peak during blowdown, then fall to lower, albeit still elevated, quasi-steady values that can persist for several hours after blowdown. Although pressures throughout the containment tend to be fairly uniformly distributed, thermal convection causes long-term temperatures in the upper containment to be generally higher than at lower levels. The resultant temperature gradients have been found to produce nontrivial global thermal stresses in the HDR steel containment. The HDR experience has been that the fictive pressure derived from pressure and thermal stresses is lower than the containment design pressure. Nevertheless, for commerical plants having steel containments, it might not be unreasonable to combine pressure and thermal loads with seismic loads in evaluating containment response, if an earthquake were postulated to occur shortly -- say within 24 hours -- after blowdown.

In addition to the magnitude of seismic loads, the deciding factors here would be (1) magnitude and duration of the post-LOCA temperature and pressure in containment, which would depend on break characteristics, and (2) the probability that an earthquake occurs during the time period of interest.

According to the generic hazard curves for the eastern and midwestern U.S. that were used in the LLNL investigations, the median probability of an earthquake larger than one SSE occurring within any given 24-hour period is about 4.1×10^{-7}, with an upper bound of about 1.4×10^{-6}.

Assuming that the probability of a double-ended break is judged to be sufficiently low so that DEGB and earthquakes can be regarded as independent random events, the following conclusions can be drawn regarding coupling of seismic and LOCA effects:

- Eliminating DEGB as a design basis event would allow pipe whip to be disregarded altogether.
- The most highly dynamic LOCA effects -- jet impingement and decompression waves -- coincide with the blowdown event; therefore, the resultant loads could be decoupled from seismic loads.
- LOCA effects in combination with seismic loads are addressed in NUREG-1061, Volume IV.

3.5 ARBITRARY INTERMEDIATE BREAKS

The position on pipe rupture postulation is given in detail in the Branch Technical Position MEB 3-1 as presented in Standard Review Plan (SRP), Section 3.6.2. This position is intended to comply with the requirements of the General Design Criteria 4, of Appendix A to 10 CFR Part 50 for the design of safety-related nuclear power plant structures and components. The rules stated in this position are intended to utilize available piping design information for postulating pipe rupture at locations having relatively higher potential for failure, such that an adequate and practical level of protection may be achieved.

Observations from many years of operating experience indicate that piping failures generally occur at high stress and fatigue locations, such as terminal ends, connections to components, elbows, reducers, T-sections, or weld joints. In those locations high stress concentrations and piping subjected to higher cyclic fatigue effects are anticipated. Most piping failures are also associated with one of many unanticipated situations for which the piping was not originally designed. Typical examples of these unanticipated situations are design, construction or operational errors, water or steam hammer, and corrosive environments. When the Branch Technical Position MEB 3-1 was developed which incorporated the operating experience data together with the advances in state-of-the-art understanding of pipe failure mechanisms.

This position requires that postulation of pipe break at various specific locations provide mechanical and environmental protection for the adjacent components and piping systems. In addition to the terminal ends, component connections and other high-potential break locations, MEB 3-1 required protection at any location in Class 1 piping where calculated stress reaches 2.4 Sm (or 80 percent of yield), or where the usage factor reaches 0.1. This requirement was initiated to account for the effect of combined stress and fatigue. For additional protection, MEB 3-1 further required that two locations be selected along the intermediate portion of the pipe even if the calculated stress and usuage factor do not exceed the specified limits. These two locations are selected at the two highest stress locations, even if their stresses are below 2.4 Sm. Similar criteria are postulated for Class 2 and 3 piping with the exception that fatigue is not a design consideration. The intent of MEB 3-1 is to obtain additional protection. As a result of these so called "arbitrary intermediate break criteria", many pipe whip restraints have been installed. These restraints have resulted in many problems, and the additional protection provided by their installation is questioned.

3.5.1 Assessment of Problems Introduced by the Arbitrary Intermediate Break Requirement

The basic intent of the arbitrary intermediate break requirement is to provide additional safety for the plant. Review of the following effects of pipe rupture protection devices leaves doubt about whether this requirement really contributes to plant safety.

Complications in Pipe System Design. Designing for the two arbitrary intermediate breaks is a difficult process, because the location of the two highest stress points tends to change several times due to the iterative process involved in the seismic design of piping systems. Although Revision 1 to the SRP (NUREG-0800, dated July 1981) provides criteria intended to reduce the need to relocate intermediate break locations when the high stress points shift due to piping reanalysis, these criteria provide little relief in practice. The actual responsibility rests on the designer who must justify that not postulating breaks at the relocated high stress points will not result in reduced safety. This requires extensive additional analyses of break-target interactions for the relocated break points and could result in design, fabrication, and installation of additional pipe whip restraints at the relocated break points as well as in the removal of previously installed restraints at superseded breakpoint locations. Furthermore, the two locations selected by the stress calculation may not be the actual locations of highest stress because the mathematical model may differ from the actual piping system. If the locations are not actually representative, proper protection may not be being provided in accordance with the system's design. The early determination of precise break locations in the piping system is important to effectively mitigate the potential consequences of a postulated break in a manner consistent with the safety significance involved.

Cost Factors. As a result of the arbitrary intermediate break requirements, an excessive number of pipe rupture protection devices have to be designed and constructed. The cost for the design, construction, and operational service and maintenance is estimated to be from $4 million for nine major systems to $30 million for all systems.

Restricted Access for In-service Inspection. In-service inspection during plant operation is a very important activity which enables the inspector to obtain early indication of a defective system. The leak-before-break concept can be implemented only when in-service inspection and/or leak detection systems provide early detection of possible cracks and potential leaks in the system. However, the pipe rupture protection devices block access to welds and thus hinder in-service inspection.

The removal and reinstallation of the pipe rupture protection devices will add to the time required to perform necessary in-service inspections. Restricted access may also increase occupational radiation exposure during repair, maintenance, and decontamination operations. The amount of additional radiation exposure typically incurred is presented in Section 6.0.

Increased Heat Loss to the Surrounding Environment. Because pipe whip restraints fit closely around the high energy piping, the piping insulation must often be cut back in these areas to avoid interferences, thus creating convection gaps adjacent to the restraints. This creates an overall increase in heat loss to the surrounding environment and is a major contributor to the tendency for many containments to operate at temperatures near technical specification limits.

Unanticipated Thermal Expansion Stress. Pipe rupture protection devices are designed not to restrict pipe-free thermal expansion. Should these devices inadvertently come into contact with the pipe itself, unanticipated stresses due to restraint of thermal expansion can be introduced. The precise consequences of this incident are difficult to assess; probabilistic analyses performed by the Lawrence Livermore National Laboratory indicate in general that the resultant reduction in flexibility reduces the overall reliability of the pipe system.

3.5.2 Proposed Resolution

Pipe rupture protection devices can introduce many negative effects on plant operations, and do not contribute to the plant safety as originally intended. Therefore, removal of the arbitrary intermediate break requirement on mechanical protection devices is warranted. Environmental qualification of equipment in the vicinity of these lines should be reviewed on a case-by-case basis until definitive criteria are developed.

The staff once considered the elimination of the arbitrary intermediate break requirement for piping systems in which stress corrosion cracking, large unanticipated dynamic loads (steam or water hammer) or thermal fatigue in fluid mixing situations could be demonstrated not to occur. After additional review, it is realized that in certain systems and for certain materials, thermal fatigue and stress corrosion cracking cannot be absolutely excluded from piping operation, nor can steam or water hammer. It may also never be possible to specify precise "acceptable levels" of thermal fatigue and stress corrosion cracking, nor to assure analytically that these levels would not be exceeded. However, if these unanticipated severe conditions were to occur, the break would most likely be located at the terminal ends, at the connections to components, and at other locations which introduce higher stress concentration or that exceed the stated threshold limits in SRP 3.6.2. These locations are not affected by relaxing this requirement. A review of pipe failure records for nuclear power plant piping up to early 1980 revealed that about 11 percent of failures occurred at intermediate locations. Further study of cases associated with piping systems having diameters greater than 4 in. indicated that most of the incidents were the results of indirect causes such as pipe being impacted by moving equipment, or pipe failure caused by support failure. As discussed in Section 9.1, these indirect causes are not associated with the piping stress calculations; therefore, current arbitrary intermediate break requirements will not be able to predict break locations. Therefore, the proposed resolution will have no negative impact.

3.5.3 Recommendations

The Task Group recommends that Standard Review Plan 3.6.2 (MEB 3-1) be revised to incorporate proposed changes eliminating the requirements for mechanical pipe rupture protection against arbitrary intermediate breaks including the development of definitive criteria related to environmental qualification of equipment as mentioned in Section 3.5.2 of this report..

REFERENCES

3.1 Letter to William J. Dircks from J. J. Ray, Chairman of the ACRS, "Fracture Mechanics Approach to Pipe Failure", June 14, 1983.

3.2 Lawrence Livermore National Laboratory. Sept. 1981. "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant". Report UCID-18967, NUREG/CR-2189, Vols. 1-9.

Vol. 1: Summary
Vol. 2: Primary Coolant Loop Model
Vol. 3: Non-Seismic Stress Analysis
Vol. 4: Seismic Response Analysis
Vol. 5: Probabilistic Fracture Mechanics Analysis
Vol. 6: Failure Mode Analysis
Vol. 7: System Failure Probability Analysis
Vol. 8: Pipe Fracture Indirectly Induced by an Earthquake
Vol. 9: PRAISE Computer Code User's Manual

3.3 Lawrence Livermore National Laboratory. "Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants". Report UCID-19988, NUREG/CR-3660, Vols. 1, 3 and 4 to be published.

Vol. 1: Summary
Vol. 2: Pipe Failure Induced by Crack Growth (August 1984)
Vol. 3: Guillotine Break Indirectly Induced by Earthquakes
Vol. 4: Pipe Failure Induced by Crack Growth, West Coast Plants

3.4 Lawrence Livermore National Laboratory. "Probability of Pipe Failure in the Reactor Coolant Loops of Combustion Engineering PWR Plants". Report UCRL-53500, NUREG/CR-3663, Vols. 1 and 3 to be published.

Vol. 1: Summary
Vol. 2: Pipe Failure Induced by Crack Growth (September 1984)
Vol. 3: Pipe Failure Indirectly Induced by Earthquakes

3.5 R.L. McNeill. "Seismic Hazard Estimates from the San Onofre Site". Letter report to J.H. Hutton, Combustion Engineering, dated September 13, 1983.

3.6 R.D. Streit. Sept. 1981. "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant: Pipe Fracture Indirectly Induced by an Earthquake". Lawrence Livermore National Laboratory, Livermore, California, Report UCID-18967, NUREG/CR-2189, Vol. 7.

3.7 For a general overview of the HDR Safety Program, see K. H. Scholl and G.S. Holman. Jan. 1983. "Research at Full Scale: the HDR Programme". Nuclear Engineering International.

4.0 RECOMMENDATIONS FOR FURTHER STAFF ACTIONS

4.1 EXEMPTION REQUESTS

As previously described in Section 3.2.5 of this report, exemptions to General Design Criterion 4 (GDC-4) of Appendix A to 10 CFR Part 50 with respect to the resolution of Unresolved Safety Issue (USI) A-2 were justified both on a technical and on a regulatory analysis basis. The Committee for Review of Generic Requirements (CRGR) observed (Section 3.2.4) after its review of the staff's topical report evaluation of the fracture mechanics analysis performed for the Westinghouse A-2 Owner's Group plants that this technology is equally applicable to other piping systems. The Pipe Break Task Group agrees with the CRGR. As enumerated elsewhere in this report, there are large safety, ORE and economic benefits that can accrue by the utilization of fracture mechanics to address the issue of piping integrity in lieu of postulating nonmechanistic accidents such as double-ended breaks of ductile piping. Therefore, the Task Group recommends that, in parallel with expedited rulemaking, the NRC continue to grant plant-specific exemptions to GDC-4 to PWR applicants and licensees who provide justification[4.1] for such requests both on a technical and safety benefit basis for their primary coolant piping. Such exemptions should relate to the requirement to postulate pipe breaks up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. Further, the scope of the exemptions only should be applicable to the measures required for protection against the dynamic effects (e.g., pipe whip, jet impingement) of postulated pipe ruptures; it should not pertain at this time to the definition of a loss-of-coolant accident (LOCA) nor its relationship to the regulations addressing design requirements for emergency core cooling system (ECCS) (10 CFR Part 50.46), containment (GDC-16, -50) and other engineered safety features.

The Task Group further believes that leak-before-break (LBB) technology has advanced sufficiently so that the use of advanced fracture mechanics technology may be applied as an alternative to the postulation of pipe breaks in other facilities and in other high energy fluid systems as defined in Standard Review Plan (SRP) 3.6.2. High energy fluid systems both inside and outside the containment may be included in the application of this technology if the recommended acceptance criteria stipulated in Section 5.0 are met. At this time, the results of a successful demonstration of LBB should be restricted to the elimination of the dynamic effects associated with postulated full flow area circumferential or longitudinal breaks in the piping. The specific dynamic effects which may be excluded are

(a) pipe whip and other pipe break reaction forces,

(b) jet impingement forces,

(c) vessel cavity or subcompartment pressurization* including asymmetric transient effects, and

(d) pipe break-associated transient loadings in functional systems or portions thereof whose pressure-retaining integrity remains intact.

The Task Group recommends that the NRC seriously consider granting exemptions to GDC-4 for this expanded application of LBB technology during the rulemaking process especially where significant benefits can be gained by doing so.

4.2 RULEMAKING

In parallel with the Task Group activities, the NRC is initiating rulemaking to preclude the need to issue exemptions permitting the use of advanced fracture mechanics (in the concept of LBB) as an alternate approach to requiring the postulation of pipe ruptures. The Task Group supports this initiative and strongly recommends that rulemaking be pursued expeditiously. The basis for this recommendation, as supported elsewhere in the value-impact and other sections of this report, is that a net safety as well as economic gain accrue from the elimination of massive protection devices in nuclear power facilities, particularly those intended to prevent whipping of ruptured

* Pressurization and environmental effects due to leakage must be evaluated.

pipes. The LBB approach can benefit licensees and applicants for operating licenses as well as applicants for future construction permits. Thus, the utility customers ultimately gain. A copy of the NRC memorandum initiating rulemaking is provided as Appendix E to this report.

4.3 DOCUMENTS POTENTIALLY AFFECTED BY RECOMMENDATIONS IN THIS REPORT

The Task Group was directed in its instructions to cite various documents that might require changes as a result of the recommendations developed in this report. The following is a citation of these documents:

Generic Issues

A-14 Flaw Detection
A-18 Pipe Rupture Design Criteria
B-16 Protection Against Postulated Piping Failures in Fluid Systems Outside Containment
No. 34 Reactor Coolant Systems Leakage.

Regulations

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Appendix A: General Design Criteria for Nuclear Power Plants.

Criterion 4 - Environmental and Missile Design Basis
Criterion 30 - Quality of Reactor Coolant Pressure Boundary
Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary
Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

Regulatory Guides

1.45 Reactor Coolant Pressure Boundary Leakage Detection Systems
1.46 Protection Against Pipe Whip Inside Containment
1.116 - Quality Assurance Requirements for Installation, Inspection,

and Testing of Mechanical Equipment and Systems

1.124 - Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports

1.130 - Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports

NUREG-0800 - Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants

3.6.1 - Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

3.6.2 - Determination of Rupture Locations and Associated Dynamic Effects

5.2.1.1 - Compliance with the Codes and Standards Rule 10 CFR Part 50.55a

5.2.1.2 - Applicable Code Cases

5.2.3 - Reactor Coolant Pressure Boundary Materials

5.2.4 - Reactor Coolant Pressure Boundary Inspection and Testing

5.2.5 - Reactor Coolant Pressure Boundary Leakage Detection.

Codes and Standards

ASME XI - IWB-3640

ANSI Draft Standard - Leak Detection.

REFERENCES

4.1 NRC Generic Letter 84-04, dated February 1, 1984.

5.0 ACCEPTANCE CRITERIA FOR LEAK-BEFORE-BREAK (LBB) SUBMITTALS

This section contains the Task Group's recommendations for application of the leak-before-break (LBB) approach in the NRC licensing process. The LBB approach means the application of fracture mechanics technology to demonstrate that high energy fluid piping is very unlikely to experience doubled-ended ruptures or their equivalent as longitudinal or diagonal splits.

The Task Group's recommendations and discussion are founded on current and ongoing NRC staff actions as presented in Section 3.0 of this report. Additional comments and discussion are presented in Appendices A and B.

Applicants and licensees who choose to justify mechanistically that breaks in high energy fluid system piping need not be postulated should provide submittals that comply with the recommended criteria in this section of the report. As a result of this justification, protection of structures, systems, and components important to safety against the dynamic effects of such postulated ruptures would not be required.

5.1 LIMITATIONS

The Task Group recommends that the following limitations apply to the mechanistic evaluation of pipe breaks in high energy fluid system piping:

(a) For specifying design criteria for emergency core coolant systems, containments, and other engineered safety features, loss of coolant shall be assumed in accordance with existing regulations, i.e., to be through an opening equivalent to twice the pipe flow area up to and including the largest diameter pipe in the system. The evaluation of environmental effects should be considered on a case-by-case basis.

(b) The LBB approach should not be considered applicable to high energy fluid system piping, or portions thereof, that operating experience has indicated particular susceptibility to failure from the effects of corrosion (e.g., intergranular stress corrosion cracking) water hammer or low and high cycle (i.e., thermal, mechanical) fatigue.

(c) For plants for which there is an operating license or construction permit, component (e.g., vessels, pumps, valves) and piping support structural integrity should be maintained with no reduction in margin for the Final Safety Analysis Report (FSAR) or Preliminary Safety Analysis Report (PSAR) loading combination that governs their design.

(d) The LBB approach should not be considered applicable if there is a high probability of degradation or failure of the piping from more indirect causes such as fires, missiles, and damage from equipment failures (e.g., cranes), and failures of systems or components in close proximity.

(e) The LBB approach should not be considered applicable to high energy piping, or portions thereof, for which verification has not been provided that the requirements of I & E Bulletin 79-14* have been met.

(f) The LBB approach described in this report is limited in application to piping systems where the material is not susceptible to cleavage-type fracture over the full range of systems operating temperatures where pipe rupture could have significant adverse consequences.

5.2 GENERAL TECHNICAL GUIDANCE

To place the above limitations in perspective and to provide guidance to potential users of the LBB approach, each step of the process required to develop the requisite technical justification for a LBB submittal is described in general terms below. A detailed description of the acceptance criteria that is listed below should be used by the staff for evaluation of each submittal follows this general discussion.

(a) Provide a discussion to support the conclusion that this piping run or system does not fall within the limitations delineated in Section 5.1.

(b) Specify the type and magnitude of the loads applied (forces, bending and torsional moments), their source(s) and method of combination.

Identify the location(s) at which the highest stresses coincident with poorest material properties occur for base materials, weldments, and safe ends.

(c) Identify the types of materials and materials specifications used for base metal, weldments and safe ends, and provide the materials properties including appropriate toughness and tensile data, long-term effects such as thermal aging and other limitations.

* "Seismic Analyses For As-Built Safety-Related Piping Systems".

(d) Postulate a flaw at the location(s) specified in (b) above that would be permitted by the acceptance criteria of Section XI of the ASME Boiler & Pressure Vessel Code. Demonstrate by fatigue crack growth analysis for Code Class 1 piping that the crack will not grow significantly during service.

(e) Postulate a throughwall flaw at the location(s) specified in (b) above. The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak detection capability when the pipes are subjected to normal operating loads. If auxiliary leak detection systems are relied on, they should be described.

(f) For geometrically complex lines or systems, performance of a system evaluation should be considered.

(g) Assume that a safe shutdown earthquake (SSE) occurs prior to detection of the leak to demonstrate that the postulated leakage flaw is stable under normal operating plus SSE loads for a long period of time; that is, crack growth if any is minimal during an earthquake.

(h) Determine flaw size margin by comparing the selected leakage size flaw (Item e) to critical size crack. Using normal plus SSE loads, demonstrate that there is a margin of at least 2 between the leakage size flaw and the critical size crack to account for the uncertainites inherent in the analyses and leak detection capability.

(i) Determine margin in terms of applied loads by a crack stability analysis. Demonstrate that the leakage-size cracks will not experience unstable crack growth even if larger loads (at least the $\sqrt{2}$ times the normal plus SSE loads) are applied. Demonstrate that crack growth is stable and the final crack size is limited such that a double-ended pipe break will not occur.

(j) The piping materials toughness (J-R curves) and tensile (stress-strain curves) properties should be determined at temperatures near the upper range of normal plant operation. The test data should demonstrate ductile behavior at these temperatures.

(k) Ideally the J-R curves should be obtained using specimens whose thickness is equal or greater than that of the pipe wall. The specimen should be large enough to provide crack extensions up to an amount consistent with J/T condition determined by analysis for the application. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques may be used if appropriate as described in Section A2.4.3 (Appendix A).

(l) The stress-strain curves should be obtained over the range from the proportional limit to maximum load.

(m) Ideally, the materials tests should be conducted using archival material for the pipe being evaluated. If archival material is not available, tests should be conducted using specimens from three heats of material having the same material specification. Test material should include base and weld metals.

(n) At least two stress-strain curves and two J-resistance curves should be developed for each of a minimum of three heats of materials having the same material specifications and thermal and fabrication histories as the in-service piping material. If the data are being developed from an archival heat of material, a minimum of three stress-strain curves and three J-resistance curves from that one heat of material is sufficient. The tests should be conducted at temperatures near the upper range of normal plant operation (e.g., 550 F). Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. One J-R curve and one stress-strain curve for one base metal and weld metal are considered adequate to determine temperature dependence.

(o) As indicated in Section 5.9.1 there are certain limitations that currently preclude generic use of limit-load analyses to evaluate leak-before-break conditions for eliminating pipe restraints. However, the Task Group believes that limit-load analysis can be used to demonstrate acceptable leak-before-break margins for the application, provided the limit moment is greater than the applied (normal operation plus safe shutdown earthquake (SSE)) moment at any location in the pipe run by a factor of at least three. Limit moment should be determined from Eq. (A-19) in Appendix A where the flow stress is determined from ASME Code minimum properties. Data obtained from future tests (see Section 10.0) may provide information that would allow less restrictive use of limit-load analyses for justifying elimination of pipe restraints.

The preceding description of the steps in performing a LBB analysis assumes that circumferentially oriented postulated cracks are limiting. If this is not the case, the analyses described in the above steps should also include the postulation of axial cracks and/or elbow cracks.

The following paragraphs of this section provide guidance for complying with the criteria described above.

5.3 STRUCTURAL INTEGRITY AND OPERATIONAL STABILITY OF PIPING SYSTEMS

For the piping run/system under evaluation, all pertinent information concerning the propensity for degradation or failure of the piping resulting from the mechanisms referred to in Section 5.1 should be provided. Operating history should be cited including system operational procedures; system or component modifications; water chemistry parameters, limits, and controls; resistance of piping material to the various forms of stress corrosion, pipe integrity under cyclic loadings, and the susceptibility of piping failures due to indirect causes.

5.4 APPLIED LOADINGS

From the stress analysis of record or other identified source for the piping run/system under evaluation, determine the highest stressed location coincident with the most limiting materials toughness properties for the loads associated with normal plant or system conditions in combination with the loads from the safe shutdown earthquake (SSE). Determine the loads, resulting from normal plant/system conditions (N) and the SSE at that location. For reference, also include a summary of loads and materials properties at other points in the system to justify the limiting location selection. If two or more locations are potentially limiting, duplicate analyses may be necessary.

At the specified location, resolve the loads into axial forces and transverse bending moments for the load cases N (normal) and N + SSE. The axial forces F, transverse bending moments, M_1 and M_2, and torsional moment, M_3, should be derived for each normal operation condition static load (pressure, deadweight, and thermal expansion). These pipe load components should be combined algebraically to define the equivalent static loads F_{1S}, M_{1S}, M_{2S}, and M_{3S}. As based on elastic SSE response spectra analyses, obtain the amplified pipe seismic loads, F_d, M_{1d}, M_{2d} and M_{3d}. Combine the static and dynamic load components as follows:

$$F = |F_s| + |F_d|$$

$$M = \sqrt{M_1^2 + M_2^2 + M_3^2}$$

where

$$M_1 = |M_{1s}| + |M_{1d}|$$

$$M_2 = |M_{2s}| + |M_{2d}|$$

$$M_3 = |M_{3s}| + |M_{3d}|$$

The preceding represents the simplest case in terms of conventional stress analyses and geometrically simple piping runs/systems. This does not preclude the use of more sophisticated stress analysis nor application to geometrically complex piping.

In the context of the loads assumed for the fracture mechanics evaluation of piping, several concerns have been raised. These are summarized below.

(1) Secondary loads should be included in loads specified under normal plant/system conditions.

(2) Consideration of "preservation of structural ductility" should be mandated (e.g., large circumferential cracks ~60 to 180° plus support failures resulting in extreme deformation).

The Task Group has the following opinions regarding the above issues. The fracture mechanics analyses described in subsequent sections of this report should include thermal expansion stresses, which are conservatively included as primary stresses. The Task Group further believes that other secondary stresses (e.g., through-the-thickness stresses) do not contribute significantly to crack driving potential and, in view of the conservative treatment of thermal expansion stresses, can be neglected.

The ductility issue apparently results from a concern that displacements during seismic events may be so large that loads on the piping system may exceed the ultimate load. It has been suggested that piping systems be shown to be able to tolerate added loads due to support failure and demonstrate that cracked sections have sufficient ductility (have net section plasticity) to absorb the energy associated with the postulated extreme displacements. These issues are also addressed in NUREG 1061, Vols. I and II.

Various means exist to demonstrate the integrity of components containing or postulated to contain flaws. For example, methods for piping include:

(1) Integrity is demonstrated if a double-ended break is not predicted for pipes containing large postulated circumferential, throughwall flaws when subjected to extreme loads that deform the pipe to the limits of geometrical restraints within the plant, or

(2) Integrity is demonstrated if a double-ended break is not predicted for pipes containing postulated circumferential, leakage-size throughwall flaws that can be detected with some margin by plant leakage detection systems during normal operation, when subjected to postulated loads that are the product of an ASME Code safety factor and the sum of normal operating and faulted loads.

Approach (1) relies on the assumption of full plasticity at the flawed pipe section and displacement-controlled loading. The pipe system is said to have adequate structural ductility if the postulated flaw extends in a stable manner (as determined using materials property data) when the pipe is bent to its physical restraints in the plant.

Approach (2) generally is applied using the elastic-plastic conditions associated with design loads for the flawed pipe section. It is typically implemented using load-controlled stresses. The pipe is said to have adequate integrity (margin against full break) if the predicted crack extension is stable with a margin under the normal operating plus faulted loads.

There can be significant differences in predictions depending on the specific analytical procedures and assumptions used in pipe integrity evaluations. Because application of each method can produce varying degrees of overall conservatism, the goal of this effort is to define a set of conditions that will ensure an acceptable degree of overall conservatism when evaluating piping integrity. While the conditions described later in this report are acceptable to the Task Group, they are not the only possible conditions that can be used. Other methods that can be shown to provide equivalent leakage and crack stability margins will also be considered on a case-by-case basis.

It is the Task Group's opinion that the conditions associated with the suggested extreme displacements do not represent a credible event. Furthermore, the suggested displacement and loading conditons are inconsistent with the ASME Code philosophy, which has been deemed acceptable for the design and operation of reactor components. Therefore, the Task Group has concluded that the loading conditions defined in the ASME Code are acceptable for performing evaluations of cracked piping. The suggested approach of postulating large displacements is acceptable but is not a requirement in the proposed guidance in this document. For consistency in the application of leak-before-break technology, users of this approach should also address each item of the general guidance of Section 5.2. The Task Group's objective is to detect any flaws or cracks in pipes by inspection or leakage while they are still relatively small.

5.5 VALID MATERIALS DATA

The ductile piping fracture mechanics analysis techniques that are applied in the leak-before-break assessment are strongly dependent on the material tensile properties and resistance to ductile crack extension. These material properties must be carefully obtained to ensure their applicability to the piping materials and operating environments of interest. Furthermore, they must be utilized in a manner consistent with the assumptions made in developing the fracture mechanics analysis techniques in order to ensure proper results. Section A2.4 of Appendix A, provides guidance for assuring the applicability of material properties data and for developing appropriate tensile and ductile fracture toughness properties for use in the fracture mechanics analyses.

5.6 CRACK GROWTH ANALYSIS

Postulate a flaw at the location(s) which had the highest stresses coincident with the most limiting materials properties for piping base materials, weldments, and safe ends. The flaw size should be no less than that which would be permitted by the acceptance criteria of the appropriate subsections of Section XI of the ASME Boiler & Pressure Vessel Code. The purpose of postulating this flaw is to demonstrate by fatigue crack growth analysis for Code Class 1 piping that the flaw will not result in a leak nor grow to critical crack size during the remaining lifetime of the plant. The fatigue crack growth analysis should be performed in accordance with the rules of IWB-3600 and Appendix A of Section XI of the ASME Boiler and Pressure Vessel Code.

5.7 SIZE OF POSTULATED THROUGHWALL FLAW

Postulate a throughwall flaw to be used in the fracture mechanics analysis at the same location(s) specified in Section 5.6. The size of the flaw should be such that the calculated leakage rate of fluids discharged from the flaw under normal operating loads should be detectable with margin. Margin should be defined both in terms of time to detect the presence of the leak and in terms of determining the magnitude of the leak (e.g., the sensitivity and the accuracy of the leakage detection system employed). The margin on the magnitude of the leakage applicable to high energy fluid system piping both within and outside of containment should be no less than a factor of 10 greater than the capability of the leakage detection systems used and adequate sensitivity and reliability of the leakage detection system should be demonstrated. The time and capability to detect the presence of a leak for fluid systems inside of containment is specified in (a), (b) and (c) below. The time and capability to detect the presence of a leak for fluid systems outside of containment will be evaluated on a system-unique basis. Note: the calculational methods for leak rate determination should be correlated with experimental data and include consideration of such effects as friction (flaw surface roughness) and 2-phase flow.

In general, for high energy fluid system piping, within containment, leakage detection systems should be demonstrated to be sufficient to provide the specified margin to detect the leakage from the postulated throughwall flaw utilizing the guidance of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems". Systems that do not measure leakage directly in gpm should be demonstrated to be sensitive to detect the leak rates by appropriate analyses or calibration. Specifically, the Task Group recommends that the specified margin can be achieved as follows:

(a) For PWRs, either operating or under construction, that meet all of the provisions of Regulatory Guide 1.45, each leakage detection system should be adequate to detect the rate of unidentified leakage, or its equivalent, of 1 gpm in less than one hour.

(b) For operating PWRs that do not meet all of the provisions of Regulatory Guide 1.45, at least one leakage detection system with a sensitivity capable of detecting an unidentified leakage rate of one gpm in four hours should be operable.

(c) For all BWRs, in accordance with the Pipe Crack Task Group recommendation, the limits on unidentified leakage should be three gpm or less and the surveillance interval should be four hours or less.

For high energy fluid system piping outside of containment, leakage detection should be demonstrated to be sufficient to provide the specified margin to detect leakage from a postulated throughwall flaw. Local leak detection devices (e.g., acoustic emission monitors, moisture sensitive tape) or other methods may be used if capable of detecting these leak rates. As stated earlier, the type of leakage detection employed and the frequency of monitoring for leakage are highly system dependent and should be evaluated on a system-unique basis.

5.8 SYSTEM EVALUATION

For geometrically complex fluid system piping (e.g., numerous fittings, pipes of various sizes) a system evaluation (refer to Section III of the ASME Boiler & Pressure Vessel Code, Appendix F) should be performed to identify changes in system response, if any, to the effects of postulated throughwall flaws and their locations. A system evaluation pertains to an assemblage of piping, components, component and piping supports, and other interconnected structures. Postulate a throughwall flaw in each size of pipe comprising the functional system. Note, these flaws are not postulated simultaneously.

The system evaluation should identify the effect of the uncracked piping on the run of pipe containing the throughwall flaw in terms of (1) fatigue analysis, (2) crack opening area and resultant leakage, and (3) stability of the flaw under the loads associated with normal operation plus the safe shutdown earthquake (SSE).

5.9 ANALYTICAL METHODS FOR THROUGHWALL CRACKED PIPES

The ability of an unflawed structure to withstand stresses resulting from applied loads is typically determined by the material's strength as determined by the materials stress-strain properties. When a flaw is introduced in the structure, the ability of the material to withstand the applied stresses may not be determined solely by the material's strength but also by its resistance to crack extension which is referred to as its fracture toughness.

If the flawed structure is fabricated from a material that has a high fracture toughness and therefore is not sensitive to the presence of a crack, the load carrying capacity of the cracked structure may still be governed by material tensile strength. In this situation an assessment of the integrity of the flawed structure can be carried out by calculating the applied stresses in the component, taking into account the flawed geometry, and comparing those stresses to a parameter related to the material's strength. For certain piping materials having high toughness (e.g., wrought stainless steel) this has been demonstrated as an effective analytical technique and techniques referred to as limit load or net section collapse analysis have been developed and validated for the evaluation of piping integrity.

If the structure of interest is fabricated from a material that has low fracture toughness and is therefore sensitive to the presence of a flaw, other analytical techniques must be used. Methods for performing fracture mechanics evaluations under linear elastic loading conditions have been extensively developed and validated. Fracture mechanics evaluations which account for the presence of substantial plastic strain are of recent development and in fact are still evolving. This section discusses the limitations and applications of the various analytical techniques to the leak-before-break evaluations discussed in this report.

5.9.1 Limit-Load Analysis

Limit-load analyses of circumferentially and axially cracked pipes have been successfully applied in many cases. Net-section collapse, plastic instability, and flow stress-dependent analyses are terms frequently used interchangeably in reference to limit-load analyses. The inherent assumption in applying such an analysis is that the material toughness is sufficient to ensure that failure loads are controlled by the material strength. Net-section collapse analysis predicts the maximum load based on the initial crack size. Hence another assumption in applying the net-section collapse approach is that there is negligible crack growth between the load where extension of the initial crack begins and the maximum load capacity of the pipe.

The net-section collapse analysis for circumferentially cracked pipe was originally(5.1) developed for applications to stainless steel pipes. The concepts were based on center cracked flat plate experiments and in subsequent research(5.2) for pipes in pure bending. The original formulation(5.1) is given in Section A2.0 of Appendix A. The efforts in Reference 5.2 showed that for the case of pure bending the ovalization of the pipe effected the limit moment, and an empirical ovalization correction function was developed. The investigations showed that the flow stress defined as 1.15 $(\sigma_y + \sigma_u)/2$ gave excellent agreement with the wrought stainless steel pipe experiments conducted at room temperature. Figure 5.1(a) shows the good agreement achieved using this flow stress representation and the ovalization correction function for wrought stainless steel material at room temperature.

For throughwall, circumferentially cracked pipe under pure axial tension stress, comparisons with existing experimental data show that the net section collapse analysis gives a good estimate of the maximum load. Figure 5.1(b) compares stainless steel and some carbon steel pipe data with net-section collapse predictions. Although the toughness of the carbon steel pipes is unknown for these experiments, it is certainly encouraging that even the 30-inch-diameter pipe tests by Kiefner(5.3) agreed reasonably well with the net-section collapse analysis.

For combined bending and pressure loading on throughwall, circumferentially cracked pipes very little data are available to assess the validity of the net section collapse approach. Comparisons of the available experimental data with net section collapse predictions are shown in Figure 5.2 for small-diameter carbon steel(5.4) and stainless steel pipes. These predictions used the principle of superposition of the net section stress contribution from axial stress and bending stress components. (This is similar to the method

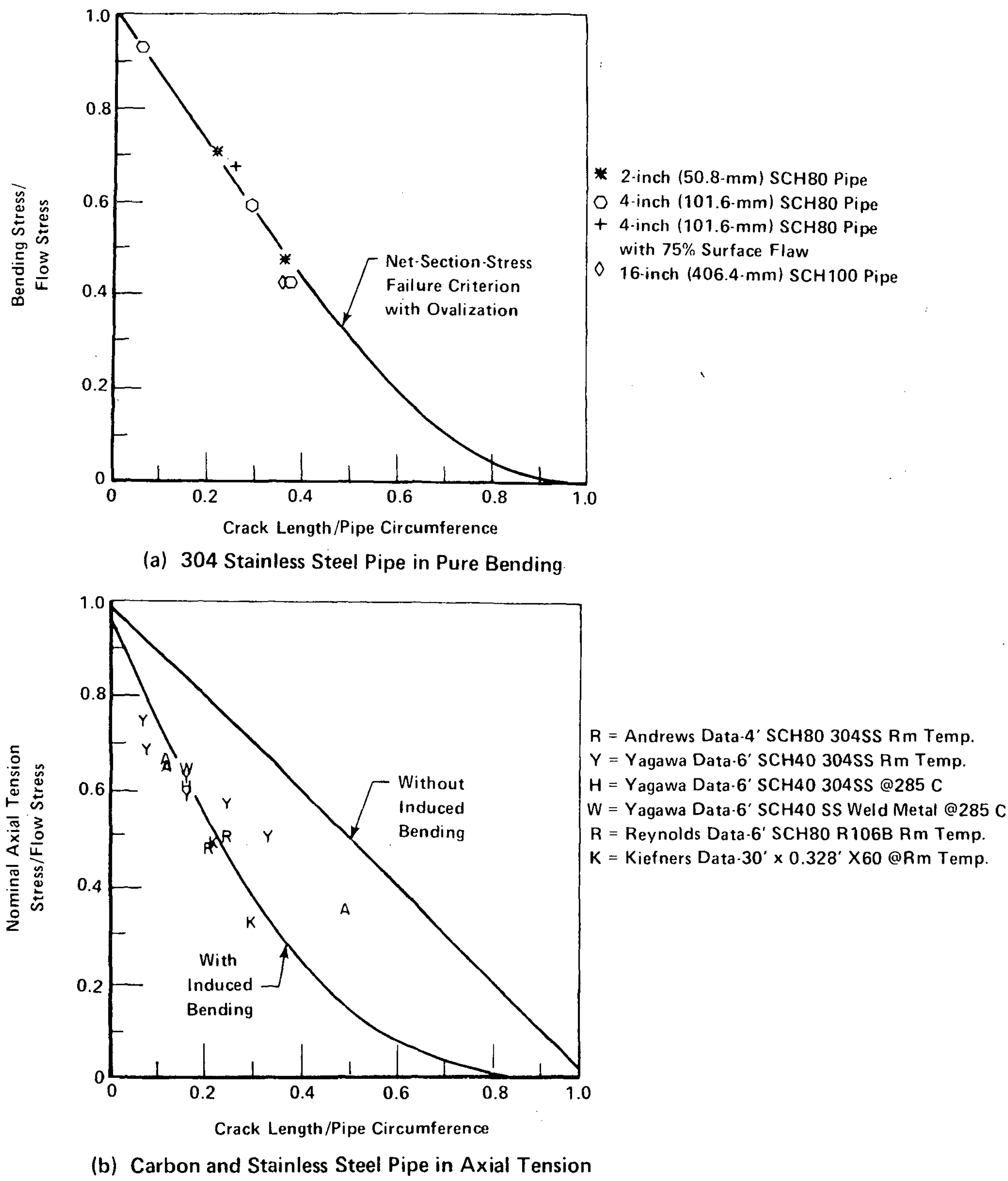
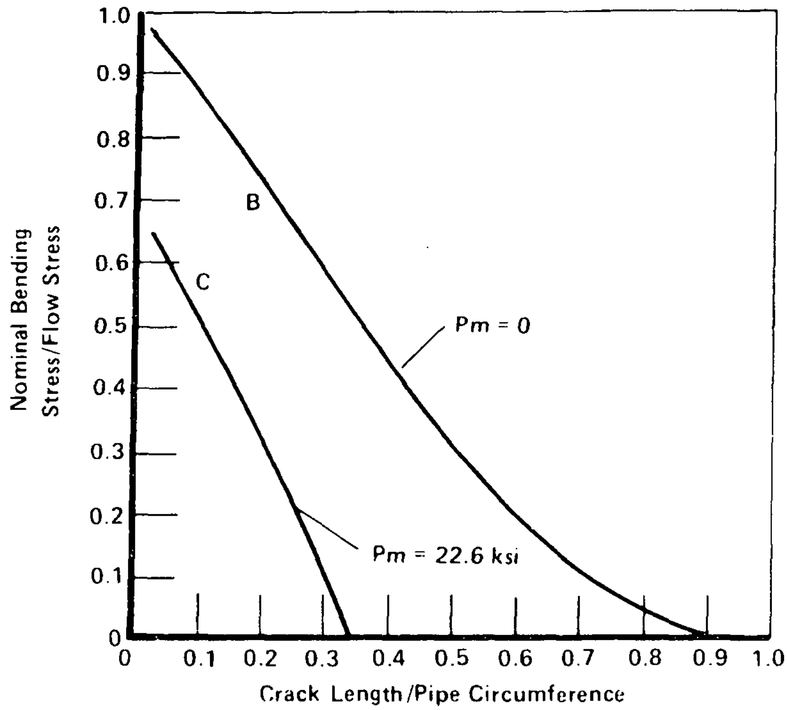


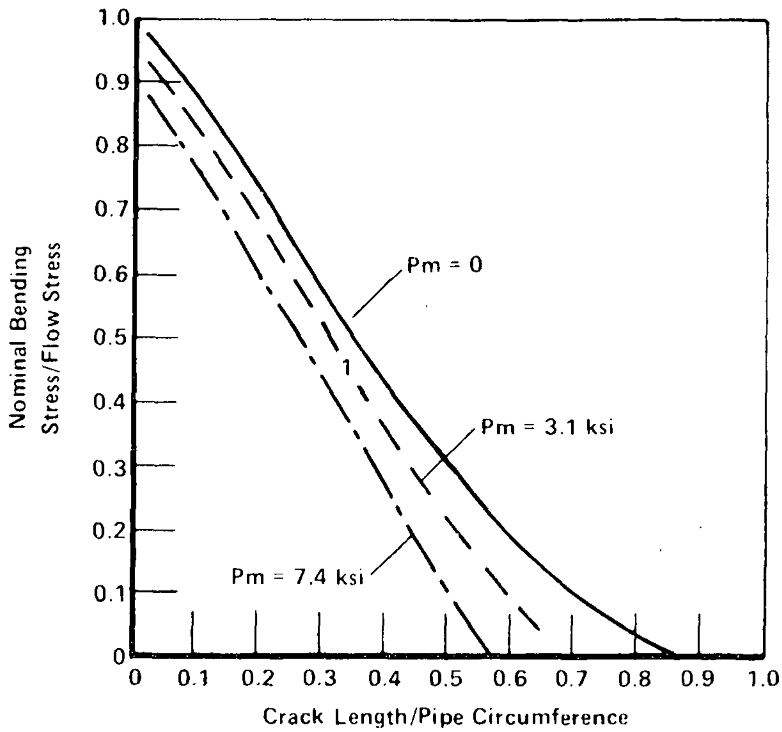
Figure 5-1 Comparisons of Experimental Data to Circumferential Crack Limit-Load Analyses



(a) Carbon Steel Pipe

Data From Reynolds-6' Dia SCH80 R106B @70 F
B = Bending Only Data, Throughwall Crack
C = Combined Bending and Pressure Data Through Crack, Axial Stress = 22.55 ksi

(a) Carbon Steel Pipe



(b) Stainless Steel Pipe

Data from Wilkowski
4" O.D. SCH80 Type 304 SS Pipe @10 C
Yield = 45.1 ksi, Ultimate = 91.9 ksi
1 = 5.25" Throughwall Crack, Pm = 3.1 ksi
2 = 3.00" Throughwall Crack, Pm = 7.4 ksi

(b) Stainless Steel Pipe

Figure 5-2 Comparison of Combined Load Net-Section Collapse Analysis to Circumferential Throughwall Cracked Pipe Data used in Appendix A, Section A3.3, for combined load J/T analyses.) For the carbon steel pipe, the predictions in Figure 5.2(a) are conservative, and for the wrought stainless steel pipe the error in predicting the bending moment was less than 5 percent.

The above comparisons are encouraging in regard to using the simple limit load analyses; however, certain limitations need to be addressed. The major concern is that for low toughness materials the limit-load approach may be nonconservative. The degree of nonconservativeness is currently unknown since tests conducted to date have been on relatively small-diameter piping and fairly high toughness materials. In addition, many previous tests were performed without measurements of crack extension; consequently it is not known if there was significant crack extension prior to maximum load. Furthermore, many previous tests were conducted using system compliances that do not appropriately model real piping compliance characteristics which may result in lower load-carrying capacity than indicated by limit load.

In the absence of experimental pipe fracture data on low toughness materials and/or representative piping system compliances, the net section collapse analysis cannot be shown to give accurate estimates of the maximum load carrying capacity for all the potential pipe cases of interest to the nuclear industry. While the toughness of most materials used in nuclear facility piping systems is considered to be adequate, there are currently a number of exceptions such as the toughness of stainless steel submerged arc weldments. The Task Group recommends that a toughness comparable to or better than that of A106 Grade B carbon steel be demonstrated to justify using the limit-load approach. The material tests recommended in Section A2.4 of Appendix A should be used to determine whether or not this criterion is met. When adequate material toughness is established it is suggested that the limit-load approach can be used when the calculated limit load is greater than the service load of interest by a factor of at least three. The use of limit-load analysis with a factor of three is intended to demonstrate that double-ended guillotine breaks will not occur. Additional applications of limit-load analysis and associated margins are applicable to other evaluations, such as that described for flaw evaluations in Volume I, NUREG-1061. For leak-before-break evaluations the flow stress should be defined as the average of the ASME code specified yield and ultimate tensile strengths at the appropriate temperature. Hence, more complicated J/T analyses and materials characterizations would not be needed for low-stressed piping systems. Future research in the NRC Degraded Piping Program will determine if this factor should be modified. It is therefore recommended that for higher stressed systems (i.e., applied stress greater than one-third of the limit moment) that a ductile fracture mechanics analysis be used. Several such analyses are described in Appendix A, but this is not to preclude the use of improved or validated methods developed in the future. As described in the next section, fracture instability under such conditions can be assessed by ductile fracture mechanics conditions which needs to account for the tearing resistance of the material and the pipe system compliance.

A final limitation on the limit load concept is that it does not provide a method to calculate leakage areas for throughwall cracked pipes. At low stresses linear elastic fracture mechanics analyses can predict the crack opening areas, but generally at stresses above half of the limit-load, elastic-plastic methods may be required.

5.9.2 Fracture Mechanics Analyses

Description of Analyses. If the piping of interest is fabricated from a material that has a low fracture toughness and is therefore sensitive to the presence of a flaw, fracture mechanics analysis techniques other than limit-load analyses should be used to evaluate structural integrity. Fracture mechanics analyses involve calculating crack driving force parameters related to the applied loading, crack size, shape and orientation and component geometries and comparing them to experimentally determined fracture resistance parameters to evaluate the integrity of the structure.

If the stresses in the cracked component are relatively low and no significant plastic deformations are predicted, linear elastic analysis techniques can be used for calculating the applied driving force parameter. Accepted methods exist for calculating this parameter. Also, limit-load analysis may be acceptable to the NRC staff as discussed in Section 5.9.1. For increasing stresses, resulting from higher applied loads or larger crack sizes, the assumption of linear elastic material response may be violated and more elaborate fracture analysis techniques are necessary. As a first approximation, plastic zone size-corrected elastic analyses may be used, but as the plasticity becomes increasingly large more sophisticated elastic-plastic fracture mechanics techniques must be used. In this case several crack driving force parameters are available, e.g., J-integral, crack opening angle, etc. Most of these parameters can be directly related to one another, and the J-integral, which is commonly used in the U.S., will be used in this discussion.

In the ductile temperature regime of interest to the analyses discussed in this report, the crack extension of interest is a materials/structural instability that depends on the change in the crack driving force, J as a function of crack extension. This change in J as a function of crack extension is represented by a parameter commonly referred to as the applied tearing modulus, T.

The J and T parameters include both elastic and plastic components, and in the absence of plastic strain degenerate to a purely elastic analysis based on well-accepted linear elastic analysis techniques. For the situation where significant plasticity exists there are several methods for calculating J. These include finite element methods and various types of closed-form estimation schemes.

The experimentally determined resistance to ductile crack extension to which the J and T parameters are compared is generally presented in the form of a J-resistance curve (the relationship between J_m and crack extension). The slope of the J-resistance curve at a given crack length and load is used to calculate a T_m value that when compared to the T value allows determination of stable ($T < T_m$) or unstable ($T > T_m$) crack extension. If no instability occurs the amount of stable crack extension can be determined by comparison of J with the J-resistance curve.

Discussion. Appendix A presents a description and evaluation of the J analysis techniques. The appendix includes an attempt to benchmark various J analysis techniques by comparing various J analysis methods with currently available experimental data that describe the moment and J values corresponding to first crack extension (see Table A-3 of Appendix A) for ferritic and stainless steel piping. The J-analyses methods evaluated include the EPRI estimation scheme,[5.5] the NUREG/CR-3464 method[5.6], and the NRC modification of the NUREG/CR-3464 method.

The results from this comparison (see Table A-4 of Appendix A) indicate that the EPRI estimation scheme is consistently conservative in predicting moment to initiation with a maximum difference of about 20 percent for ferritic piping and 30 percent for stainless steel piping. The method described in NUREG CR-3464 was consistently nonconservative in predicting moment to initiation with a maximum difference of about 10 percent for stainless steel and 20 percent for ferritic piping. The NUREG approach does not account for strain hardening of piping materials. In an attempt to remedy this situation, the NRC staff modified the kink angle equations in the NUREG document to include a strain hardening term based on the Ramberg-Osgood equation. Otherwise, the staff analytical model follows the procedures of NUREG/CR-3464. While the staff modification improves the correlation between calculated and experimental results, it still appears to be somewhat nonconservative. The modified procedure predicted a maximum difference of about 10 percent overprediction for stainless steel and 20 percent overprediction for ferritic steel.

The EPRI estimation scheme consistently overpredicted the value of J at the experimental initiation moment. The computed J values differed by a maximum factor of seven for 16-in.-diameter stainless steel pipe and three for the ferritic pipe. The NUREG/CR-3464 estimation method consistently underpredicted the value of J at initiation. The computed J values differed by a maximum factor of 10 for stainless steel pipe (4-in.-diameter) and 4 for ferritic pipe. The NRC-modified NUREG method underpredicted J in the majority of cases. The computed J values were underpredicted by a maximum factor of three for both the stainless steel pipe (4-in.-diameter) and the ferritic pipe. The comparisons with limited data assessing the effect of pipe diameter showed that the EPRI analysis became more conservative for larger diameter pipe. From comparisons to limited combined bending and pressure pipe fracture experimental data, the degree of conservatism in the EPRI estimation scheme was found to be the same as for the benchmark comparisons to pipe fracture data for pure bending. The above comparisons assume that the reported experimental values of J and moment at crack initiation are correct. In recognition of the fact that there probably are experimental uncertainties associated with these data, the Task Group also estimates the degree of scatter by plotting the reported data points against the averages for the seven 8-in.-diameter ferritic steel pipe tests identified in Table A-3 of Appendix A. This was done for both the J ratios (Figure A-7) and moment ratios (Figure A-8) at crack initiation. From these figures in Appendix A, it can be seen that there apparently are uncertainties due either to experimental measurement or limitations to the analyses. The experimental scatter band is shown as the box in each of the figures. While the differences among the three estimation procedures discussed above maintain the same trends relative to one another, the NRC-modified NUREG approach appears, on the average, to best fit the experimental data when consideration is given to potential material property and experimental measurement uncertainties. Additional experiments, especially with larger carbon steel and stainless steel pipes, and refinements to the analytical procedures are deemed to be desirable to resolve this issue.

The above results indicate that there can be significant computational differences between the existing estimation schemes. The Task Group believes that the computational uncertainty is appropriately accounted for by the margins specified in Section 5.2. However, the analyst should take steps to ensure that significant nonconservative predictions are not made.

5.9.3 Recommendations

Based on the discussion in Sections 5.9.1 and 5.9.2 and Appendix A the Task Group has the following recommendations regarding application of ductile piping fracture mechanism to leak-before-break evaluations.

(a) As indicated in Section 5.9.1 there are certain limitations that currently preclude generic use of limit load analyses to evaluate leak-before-break conditions for eliminating pipe restraints. However, the Task Group believes that limit load analyses can be used to demonstrate acceptable leak-before-break margins for the application provided the limit moment determined from Eq. (A-19) in Appendix A is greater than the applied (normal operation plus SSE) moment at any location in the pipe run by a factor of at least three. Data obtained from future tests (see Section 10.0) may provide information that would allow less restrictive use of limit load analyses for justifying elimination of pipe restraints.

(b) When crack extension is predicted to occur, stability analysis should be performed (see Section A3.4 in Appendix A) to determine if adequate margins against crack instability are maintained. Stability computations should include crack extension characteristics of the materials as defined by appropriate J-R curve data.

(c) Ideally, the materials tests should be conducted using archival material for the pipe being evaluated. If archival material is not available, tests should be conducted using specimens from three heats of material having the same materials specification. Test material should include base and weld metals.

(d) At least two stress-strain curves and two J-resistance curves should be developed for each of a minimum of three heats of materials having the same material specifications and thermal and fabrication histories as the in-service piping material. If the data are being developed from an archival heat of material, a minimum of three stress-strain curves and three J-resistance curves from that one heat of material is sufficient. These tests should be conducted at temperatures near the upper range of normal plant operation (e.g., 550 F). Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe rupture could have significant adverse consequences. The tests at lower temperatures are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. One J-R curve and one stress-strain curve for one base metal and weld metal at the lower temperature are considered adequate to determine temperature dependence. The stress-strain curves should be obtained over the range from the proportional limit to maximum load.

(e) The J-R curves should be obtained using specimens whose thickness is equal or greater than that of the pipe wall. If possible, specimens should be large enough to provide crack extension up to an amount that allows the J/T analysis to be performed without extrapolation of the J-resistance curve. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques may be used if appropriate as described in Section A2.4 of Appendix A.

5.10 MARGIN TO CRITICAL CRACK SIZE

As stated in Section 5.2(h), using normal plus SSE loads, it should be demonstrated that there is a margin of at least two to account for the uncertainties inherent in the analysis and in the capabilities of leakage detection systems. The factor of two stems from the equivalent factor of the $\sqrt{2}$ on stress intensity for flaw evaluation under Level D loadings in IWB-3600 of Section XI of the ASME Boiler & Pressure Vessel Code.

5.11 MARGIN ON LOADS

In general, the loads specified for the design of nuclear facility piping are upper bounds of loads actually experienced. However, it is desirable to have an estimate of the actual load at which the leakage size flaw might experience unstable growth in recognition of the fact that there are materials property and analytical procedure uncertainties even if the loads are reasonably well known. A load margin of at least the $\sqrt{2}$ times the normal plus SSE loads is recommended at this time.

5.12 MARGINS IN GENERAL

The Task Group recommends that the overall LBB approach should be implemented conservatively. It is recognized, however, that there are various ways in which conservatism can be incorporated and that large margins are not necessary in each step of the process provided that the overall objective is met. Thus, the specific margins recommended in the previous paragraphs could be modified provided that equivalent conservatisms are included elsewhere in the LBB approach. It is the Task Group's opinion that the NRC staff should have the flexibility to use engineering judgments on a case-by-case basis.

5.13 SUPPORT MARGINS

As specified in Section 5.1, for plants with operating licenses or construction permits, component and piping supports should maintain the same margins that currently control their design for structural integrity. Margins for component and piping support structural integrity for applicants for a construction permit will be determined in the near future.

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6.0 VALUE-IMPACT

This section discusses value-impact considerations associated with the elimination of double-ended guillotine break (DEGB) as a design requirement. A detailed assessment of value-impact for a proposed regulation change cannot be made without first knowing what the change -- and the related change in plant risk -- will be. However, a review of value-impact assessments performed for specific break issues offers insight into the potential implications of changes in pipe break criteria.

6.1 RESOLUTION OF USI A-2

A value-impact assessment was performed for 16 Westinghouse PWR plants affected by USI A-2 asymmetric blowdown loads resulting from DEGB at specific locations in reactor coolant loop piping.[6.1] These locations include the reactor pressure vessel (RPV) nozzle-pipe interface in the reactor cavity plus other selected break locations external to the reactor cavity. These postulated ruptures could cause pressure imbalance loads, both internal and external, to the primary system which could damage primary system equipment supports, core cooling equipment, or core internals, and thus contribute to core melt frequency. The results of this assessment are summarized in Table 6-1.

6.1.1 Value

The estimated reduction in public risk for installing additional pipe restraints and modifying equipment supports as necessary to mitigate or withstand asymmetric pressure blowdown loads is very small, only about 3-1/2 man-rem total for the nominal case for all 16 plants considered. Similarly, the reduction in occupational exposure associated with accident avoidance due to modifying the plants is estimated to total less than 1 man-rem. These small changes result from the estimated small reduction in core melt frequency of 1×10^{-7} events/reactor-year that would result from modifying the plants.

Table 6-1 Results of USI A-2 Regulatory Analysis

(Leak-Before-Break Value-Impact Summary (Total for 16 Plants))

Factors	Dose (man-rem)			Cost ($)		
	Nominal Estimate	Lower Estimate	Upper Estimate	Nominal Estimate	Lower Estimate	Upper Estimate
Values (man-rem)						
Public Health	-3.4	0	-37	-	-	-
Occupational Exposure (Accidental)	-0.8	0	-30	-	-	-
Occupational Exposure (Operational)	$+1.1x10^4$	+3500	$+3.2x10^4$	-	-	-
Values Subtotal	$+1.1x10^4$	+3500	$+3.2x10^4$	-	-	-
Impacts ($)						
Industry Implementation Cost[a]	-	-	-	$-50x10^6$	$-25x10^6$	$-75x10^6$
Industry Operating Cost	-	-	-	$-6.5x10^5$	$-3.3x10^5$	$-9.8x10^5$
NRC Development and Implementation Cost[b]	-	-	-	$-4.0x10^5$	$-2.0x10^5$	$-6.0x10^5$
Power Replacement Costs	-	-	-	$-60x10^6$	$-30x10^6$	$-90x10^6$
Public Property	-	-	-	$+2.4x10^4$	0	$+2.6x10^6$
Onsite Property	-	-	-	$+1.5x10^4$	0	$+4.6x10^6$
Impact Subtotal	-	-	-	$-110x10^6$	$-55x10^6$	$-165x10^6$

(a) Does not include industry costs expended to date to prepare plant asymmetric pressure load analyses and pipe fracture mechanics analysis.

(b) Does not include NRC cost expended to date to develop issue (NUREG-0609) and to evaluate Westinghouse pipe fracture mechanics analysis.

However, the occupational exposure estimated for installing and maintaining the plant modifications would increase by 11,000 man-rem. Consequently, the savings in occupational exposure by not requiring the plant modifications far exceed the potentially small increase in public risk and avoided accident exposure associated with requiring the modifications.

6.1.2 Impacts

The estimated industry costs to install plant modifications to withstand asymmetric pressure loads is about $50 million. Estimated power replacement costs would be an additional $60 million since the plant modifications would be extensive and involve working in areas with limited equipment access and significant radiation levels so that the work would probably extend plant outages beyond normally planned shutdowns. It is estimated that maintenance and inspection of the modifications for the remaining life of all the plants would cost $650K to $1 million. The cost for recalibrating leak detection systems is estimated at about $350K. The above costs do not include the industry costs expended to date to perform asymmetric pressure load analysis and fracture mechanics analysis; these costs are considered small compared to the plant modification and power replacement cost indicated above. It would cost NRC approximately $800K in staff review effort if plant modifications to withstand asymmetric pressure loads were to be installed. If they are not installed and this cost is saved, it is estimated that NRC cost would be $400K to review leak detection system calibration work and plant technical specification revisions. Exempting the plants from installing modifications would result in a net saving of $400K in NRC costs. Consequently, the savings, both in terms of occupational radiation exposure and costs, far outweigh any potential benefits (e.g., decrease in public risk and avoided accident exposure) from plant modifications.

6.2 CONCLUSIONS

Any detailed value-impact assessment of changes in pipe break criteria requires knowledge beforehand of the specific changes themselves. Nevertheless, a review of assessments already completed for specific pipe break issues implies the following general conclusions for elimination of DEGB as a design basis for PWR reactor coolant loop piping:

(a) Elimination of pipe whip restraints would only negligibly increase public and occupational radiation exposure (ORE) resulting from pipe break accidents.

(b) Elimination of pipe whip restraints would improve access to pipe welds for in-service inspection (ISI), and thereby significantly reduce ORE during inspection. Improved access would also reduce ORE during normal plant maintenance, although to a lesser degree. This benefit would apply both to operating plants and to plants under construction.

(c) For operating plants not already having pipe whip restraints, eliminating DEGB -- and thus the need to install restraints -- would reduce ORE during installation. For the sixteen A-2 plants, installation and maintenance of these restraints would avoid the occurrence of ORE by about 11,000 man-rem compared to a small increase in public risk and accident avoided occupational exposure of less than 5 man-rem.

REFERENCES

6.1 NRC Generic Letter 84-04, Enclosure 2, "Regulatory Analysis of Mechanistic Fracture Evaluation of Reactor Coolant Piping: A-2 Westinghouse Owners Group Plants", February 1, 1984.

7.0 INDUSTRY INITIATIVES

Various segments of the utility industry have pursued avenues leading to relief of the NRC criteria on pipe break, jet loads, and pipe whip. This section presents an overview of fairly recent correspondence pertaining to these matters. The Atomic Industrial Forum (AIF) generally has served as the spokesman and coordinator for the nuclear utility industry.

7.1 NUCLEAR INDUSTRY CORRESPONDENCE

The AIF, under the Committee on Reactor Licensing and Safety (CRLS), and more specifically, within the Subcommittee on Load Combinations, has been pursuing the subject of load combinations in the design of nuclear piping systems for several years. Earlier exchanges in correspondence with NRC occurred in 1978 and 1980 culminating in a letter[7.1] from Murray Edelman, Chairman, CRLS, to Harold Denton dated March 28, 1983. Specific criteria with regard to pipe breaks were suggested in this letter. A letter[7.2] from Harold Denton to Murray Edelman encouraged further industry suggestions. This culminated in the transmittal of pipe break criteria developed by the AIF Subcommittee from Murray Edelman to Harold Denton. Table 7-1 is an attempt to consolidate the information contained in the pipe break criteria onto one page. Finally, the AIF, in a document entitled "Industry Initiatives", transmitted an overall position with regard to pipe break including a value-impact analysis discussed in Section 7.2. This document is presented in Appendix C-1.

Table 7-1. Tabulation of Pipe Break Criteria[a]

Pipe Break Criteria	Primary System	Other Piping Inside Containment	Other Piping Outside Containment	Non-ASME XI Piping
Break Assumption	Need not assume mechanistic axial or circumferential (longitudinal vs guillotine) breaks.	Intermediate breaks not considered.[b] Retain circumferential breaks at terminal ends per MEB 3-1 unless detailed fracture mechanics justifies dropping.[c] If DEPB is considered, loads are considered.	Intermediate breaks not considered.[b] (Same as Inside Containment)	If criteria of MEB 3-1 are exceeded, design to accommodate DEPB at terminal ends & intermediate location for seismically designed high energy piping.[d] if non-seismic design, assume breaks at each tee, elbow, fitting per current criteria. Consider environmental effects of such breaks.
Dynamic Effects (local pipe whip, reactor loads, jet impingement, reactor cavity pressurization, subcompartment pressurization, pump overspeed, reactor internals dynamic loads, dynamic loads on piping attached to primary system).	Need not be considered.			
System Design Break; to design ECCS, Containment, etc.	Use DEGB.	Not stated. (Primary controls)	For environmental effect & equipment qualifiction, either: 1) new CPs use current regulatory criteria, or crack $w = t/2$, $\ell = d/2$ in each compartment containing high energy piping; or 2) new design, crack $w = t/2$, $\ell = d/2$.	Use $w = t/2$ and $\ell = d/2$.
Support Loads (normal operation + SSE)	<75% of OP_N + SSE + DEGB.	Use existing rules, or < 75% OP_N + SSE + DEGB.	Same as Inside Containment.	
Alternate Break Size	None given.	Use DEPB if susceptible[b] per MEB 3-1 where stress criteria exceeded.	Same as Inside Containment.	
Simultaneous SSE + Pipe Break for Design Consideration	Inferred that it is not used, but not explicitly stated.	Explicitly stated not to be used.	Same as Inside Containment.	

(a) Source: Abstracted from AIF proposal to NRC, Edelman to HR Denton, 7/14/83, "Pipe Break Criteria."

(b) Unless susceptible to corrosion, thermal fatigue, water hammer, etc.

(c) If fracture mechanics confirms longest stable crack under OP_N + SSE is > 2 times the size of a crack that leaks at 5X the level of detectability, it is acceptable.

(d) If structures through which seismically designed piping passes are not seismically analyzed, it is OK if analysis demonstrates the structure will not collapse.

R. P. Schmitz, Chief Nuclear Engineer of Bechtel Power Corporation, expressed views similar to those of AIF in a letter to Richard Vollmer of NRR.[7.4] This letter cited cost benefits, accessibility, and other factors related to a pipe break. A paper by Mr. Schmitz entitled "Proposed Changes in Intermediate Pipe Break Criteria"[7.5] was presented at a Committee on the Safety of Nuclear Installation's (CSNI) meeting on leak before break. This paper is Appendix C-2 of this report. It deals with various aspects including cost benefit.

W. H. Owen, Executive Vice President of Duke Power Company, in a letter to William Dircks, dated September 19, 1983, made specific requests for relief from certain pipe break criteria at McGuire and Catawba stations.[7.6]

Harold Denton responded to the letter from W. H. Owen[7.7] and cited specific conditions to be met. Further correspondence culminated in a specific safety evaluation for Catawba.[7.8]

Table 7-1 presents an overview of the AIF position and, not too surprisingly, covers the points made by Schmitz and Owen. In essence, Table 7-1 appears to represent an accumulation of the industry positions.

7.2 INDUSTRY PERSPECTIVE ON VALUE-IMPACTS

Section 6.0 contains the NRC value-impact analysis. This section (7.2) presents a distillation of industry comments pertinent to value-impact.

The two sources of value impact information are contained in Appendices C-1 and C-2. The AIF report cites general figures only such as

- design, procurement, and construction costs related to pipe rupture hardware ($20M to $40M/unit),
- number of pipe whip restraints in a "typical" light water reactor (250 to 400/plant), and
- installation times, including design, etc. (150,000 to 250,000 man-hours).

Obviously the preceding figures will vary from plant to plant, depending on its status, whether being designed, in early stages of construction, approaching a near-term operating license (NTOL), or operating.

Another factor of some concern is occupational radiation exposure (ORE). Values cited in Appendix C-1 are in terms of several hundred man-rem.

Appendix C-2 contains values reported by Bechtel that are similar to those in Appendix C-1, namely,

- design analyses, materials, construction ($100,000/restraint),
- number of restraints on a typical plant (300),
- overall costs/nuclear unit ($30M-$50M), and
- manpower per typical plant for overall installation of restraints (250,000 man-hr/500-600 MWe LWR).

REFERENCES

7.1 Letter Murray Edelman, Chairman AIF Committee on Reactor Licensing and Safety, to Harold Denton, NRR, dated March 28, 1983.

7.2 Letter Harold Denton, NRR, to Murray Edelman, AIF, dated May 2, 1983.

7.3 Letter Murray Edelman, AIF, to Harold Denton, NRR, dated July 14, 1983, transmitting AIF "Suggested Pipe Break Criteria".

7.4 Letter R. P. Schmitz, Bechtel, to R. H. Vollmer, NRR, dated June 16, 1983.

7.5 R. P. Schmitz, "Proposed Changes in Intermediate Pipe Break Criteria", presented at CSNI meeting on Leak Before Break, September 2, 1983, Monterey, Ca., NUREG/CP-0051, August 1984.

7.6 Letter W. H. Owen, Duke Power Company, to W. J. Dircks, NRC, dated September 19, 1983.

7.7 Letter H. R. Denton, NRR, to W. H. Owen, Duke Power Company, dated October 17, 1983.

7.8 Catawba Nuclear Station Unit 2, "Safety Evaluation for the Elimination of Arbitrary Intermediate Pipe Breaks", March 20, 1984, Docket 50-414.

8.0 FOREIGN REGULATORY REQUIREMENTS

As will be noted in this section, most countries followed the lead of the U.S. Atomic Energy Commission (AEC) with regard to pipe breaks. The DEGB was generally accepted. In fact, Section 8.2 indicates that most countries retain this criterion. The following section (8.1) discusses the exception--the Federal Republic of Germany--which initially used DEGB and made a change quite recently.

8.1 FEDERAL REPUBLIC OF GERMANY

Early German reactors such as KRB (Gundremmingen) and KWL (Lingen) based containment design on the instantaneous rupture of a major line. In the case of KRB, it was one of the recirculation lines. At KWL, it was the largest pipe in the reactor primary system. This represented the situation in the early 1960s.

Subsequently, extensive experimental and analytic work was conducted that served as a basis for a relaxation of the original criteria. Two papers presented at an IAEA symposium[8.1,8.2] presented the experimental and analytic bases for the changes in pipe break criteria. The experimental study covered extensive testing of flawed vessels representative of primary system piping. The conclusions of the experimental study are repeated below.[8.1]

- Vessel with longitudinal flaw. If the results obtained from vessel failure tests are transferred to real components, which is permissible without restriction because of the test dimensions and conditions selected, the implications of the tests for the primary piping system fracture hypothesis, in terms of the 'basis-safety approach', can be summarized as follows:

  - 'Basis-safety' rules out catastrophic failure of the pressure boundary components in regular operation and in postulated emergencies; The leak-before-failure criterion is validated for the whole upper shelf impact energy region from 30 J to over 100 J;

- In view of the assurances provided by 'basis-safety'--conservative limitation of stresses and increased stringency of toughness requirements--critical crack lengths can be ruled out. A leakage due to a crack can occur only as a small local gap ($\leq$ 0.1 F).

- Vessel with circumferential flaw. To the extent that the results are applicable to primary piping systems, the load and fracture behavior of the tested vessels with circumferential flaws can be summarized as follows:

  - Catastrophic failure can occur under internal pressure loads at the level of the operating pressure for primary piping systems only if the flaw is improbably long and deep;

  - Assuming a constant nominal stress level, the critical flaw length becomes shorter with increasing bending moment. Failure in the form of leakage can occur only with high bending moment and very deep flaws.

The analytic studies of Reference 8.2 cover the fracture mechanics analyses serving as a basis for the justification of leak before break for axial and circumferential flaws. As can be seen from the summary below, they rule out catastrophic failures.[8.2]

The basis-safety concept ensures a quality standard of the reactor coolant pressure boundary which precludes a catastrophic failure.

Ruptures need not be postulated for the main coolant line because of the reduction in stresses (e.g., by optimizing the mechanical design), the extremely tough condition of the materials and the high quality of manufacture and processing.

The "bottom line" is the acceptance or rejection of a suggested position by the regulatory organization of a given country. Two documents[8.3,8.4] are available concerning the modification of the FRG position on pipe break. The first[8.3] discusses the bases for the changes in the guidelines of the Reaktorsicherheit Kommission (RSK). The second[8.4] presents the RSK guidelines for PWRs with regard to postulated leaks and breaks, and postulated leak cross sections in the main coolant pipe whip restraints. This was extended to both the main steam and feedwater lines inside containment. In the case of BWR systems, the replacement of austenitic piping permitted similar decisions.

Reference 8.4 covers the explicit details and they are repeated below. As can be seen, the design pipe break is 10 percent of the pipe cross section. The apparent basis for this figure is a back calculation to establish levels of jet forces that can be handled without massive restraints.

With regard to the original DEGB, it is still used as it was originally by the AEC. Both containment and ECCS are designed on the basis of a DEGB (200 percent of cross section).

Since the FRG positions relate to actions suggested in this report, they are repeated here with the exception of a footnote not relevant to issue.

... "(1)Reaction and jet forces acting on pipes, components, component internals, and buildings.

1. Concerning the load assumptions for reaction and jet forces on pipes, components, component internals, and buildings, a spontaneously opening leak (linear opening behavior, opening time 15 ms) with a cross section of 0,1 F (F = open cross section) shall be postulated for different break positions.

2. In order to cope with the consequences (pressure increase in the reactor pit, release-pressure-wave acting on the reactor pressure vessel internals) of a postulated leak with a cross section of 0,1 F between the reactor pressure vessel and the biological shield, measures shall be taken, e.g., double pipes in the area of the main coolant pipe penetrations through the biological shield.

2. Presumptions for the design and the safety demonstration of the emergency core cooling systems, the containment vessel and its internals as well as the supports of the reactor coolant system components.

For the design and examination by calculation the following postulates are relevant:

1. The analysis of the emergency core cooling efficiency (reference to Section 22.1.1) shall be based on leak cross sections in the main coolant pipes up to 2 F. The emergency core cooling systems shall be designed accordingly.

2. The determination of the containment vessel design pressure as well as the determination of pressure differences inside the containment vessel shall be based on leak cross sections up to 2 F.

   The determination of design pressure and design temperature for incident resistant electrical equipment shall be based on leak cross sections of 2 F as well.

3. For the demonstration of stability of components, reactor pressure vessel, steam generators, main coolant pumps, and pressurizer, the following assumptions shall be made:
   The stability of the components shall be assured for a static force P_{ax} with

   magnitude: $P_{ax} = p \times F \times S$
   P = nominal operation pressure
   F = open cross section
   S = 2 (safety margin)

   origin of force: Middle of the pipe cross section in the area of the nozzle circumferential weld direction of force: Middle line of the nozzle acting towards the components.

This force acts only on one nozzle for the time being. The stability shall be demonstrated for each nozzle separately.

NOTE: With respect to the steam generator, the stability shall be assured for the connection to the secondary circuit in the same way.

(3) Deterministic postulated leak cross section in the reactor pressure vessel.

1. In view of the restraints of the reactor pressure vessel, the stresses acting on the reactor pressure vessel internals and the design of the emergency core cooling systems, a leak of about 20 cm^2 (geometric cross section: circular) shall be also postulated below the reactor core upper edge. Prior defects of the reactor pressure vessel which might lead to a leak size of more than 20 cm^2 shall be detectable in time by means of suitable monitoring measures.

2. The design shall also be based on the consequences of the sudden break of a control assembly nozzle involving the maximum possible leak cross section as well as the postulated leaks in the reactor pressure vessel.

(4) Pressure barrier of the low-pressure system towards the high-pressure system.

Provisions shall be made against pressurizing of the low-pressure system as a result of a failure of the pressure barrier towards the high-pressure systems (pressure-retaining boundary). The provisions may include recurrent tests of valve functions, measurements of the pressure between two successive valves and the indication of leaks in the control room.

21.2 Postulated Leaks and Breaks in the Main Steam and/or Feedwater Pipe

(1) The loads acting on the steam generator heating tubes as a result of the static and transient stresses (pressure-surge, flow forces, static pressure differences along the steam generator heating tubes) in case of a main steam and/or feedwater pipe break or remaining open of a secondary safety valve shall be determined. It shall be demonstrated that the steam generator heating tubes will cope with these stresses. In principle, however, in this incident analysis the failure of a few steam generator heating tubes shall be postulated as a single failure which shall be considered by the assumption of a total break (2 F) of a steam generator heating tube in the concerned steam generator comprehensively. For the case of a main steam pipe break outside the outer isolating valve an additional 'isolating valve nonclosure' single failure, a steam generator heating tube failure need not be postulated if the above demonstration has had a positive result.

(2) The effects of a main steam pipe break and of a cold water transient on the reactivity behavior and on the changes in pressure and temperature in the reactor as well as the resulting stresses on the reactor pressure vessel and its internals shall be kept under control....

8.2 OTHER COUNTRIES' PIPE BREAK REQUIREMENTS

Previously in Sections 2.1 and 8.0, the original bases for pipe break were discussed. Essentially all countries operating LWRs used the AEC DEGB criteria. A paper in 1967 by Vinck and Maurer[8.5] discussed maximum hypothetical accidents used for containment design and for establishing the radiological consequences of such an accident. The early plants within Euratom were discussed in the context of the MHA. Specific plants cited were SENN, KRB, KWL, GKN, SENA, SELNI and KWO. All used the instantaneous rupture of a major pipe as noted:

SENN - Instantaneous rupture of largest primary pipe

KRB - Instantaneous rupture of a recirculating line

KWL - Instantaneous rupture of largest primary pipe

GKN - Main steam line rupture inside drywell

SENA - Instantaneous rupture of primary and secondary loop

SELNI - Instantaneous rupture of largest primary pipe

KWO - Instantaneous rupture of largest pipe.

In essence, these criteria are still applied in Europe with the exception of Germany.

Two papers[8.6,8.7] cite the Japanese criteria which are similar to the original AECs. Reference 8.6 cites three plants (Tsuruga, Fukushima, and Mihama) where the maximum hypothetical accident (MHA) was the instantaneous break of the largest pipe in the reactor coolant recirculation loop or the reactor primary system. In addition, they analyzed the rupture of the main steam line.

In a later session of the Geneva Conference,[8.7] Ando discussed maximum credible accidents used in Japan. These are cited below.

	Serious Accident	Hypothetical Accident
BWR:	(1) Main stream pipe rupture (2) Loss of coolant (3) Rupture of off-gas storage tank	Ditto, neglecting the effect of ECCS, to consider 100 percent fuel melt.
PWR:	(1) Loss of coolant (2) Piping rupture in the steam generator	Ditto, neglecting the effect of ECCS, to consider 100 percent fuel melt.

As can be seen, the original AEC pipe break criteria were accepted universally by other countries having LWRs. It is our understanding that other countries are reviewing the pros and cons of more relaxed pipe break criteria similar to those in Germany. To our knowledge, no positive action is expected from any country in the near future.

It was reported at a recent Committee on the Safety of Nuclear Installations (CSNI) meeting in San Antonio (6/21 - 6/22/84) that the Italian Regulatory Authority (ENEA) had adopted a position essentially the same as that of the FRG.

REFERENCES

8.1 K. Kussmaul, W. Stoppler, D. Sturm, and P. Julisch. "Ruling-out of Fractures in Pressure Boundary Pipings, Part 1: Experimental Studies and Their Interpretation". pp. 211-235, MPA-Stuttgart, FRG, IAEA-SM-269/7.

8.2 G. Bartholome, W. Kastner, E. Keim, and R. Wellein. "Ruling-out of Fractures in Pressure Boundary Pipings, Part 2: Application to the Primary Coolant Piping", pp. 237-254. Kraftwerk Union AG, Erlangen, FRG. IAEA-SM-269/7.

8.3 H. Schulz. "Current Position and Actual Licensing Decisions on Leak Before Break in the Federal Republic of Germany". (GRS) mbH, Koln, FRG. Presented at CSNI Specialist Meeting on Leak Before Break, Monteray, CA, 9/1-2/83, NUREG/CP-0051, August 1984.

8.4 "RSK Guidelines for Pressurized Water Reactors", 3rd Ed., 10/81. In (GRS) mbH Translations - Safety Codes and Guides. Edition 5/82.

8.5 W. F. Vinck and H. Maurer. "Some Examples of the Relationship Between Containment and Other Engineered Safeguard Requirements, Accident Analyses and Site Conditions". In Proceedings of a Symposium on the Containment and Siting of Nuclear Power Plants, pp. 383-402. EURATOM, Brussels, IAEA-89/58.

8.6 H. Osawa and Y. Togo. "Application of 'A Guide to Reactor Site Evaluation' in Japan". In Proceedings of a Symposium on the Containment and Siting of Nuclear Power Plants, IAEA, Vienna, 1967, pp. 19-33, IAEA SM-89/25.

8.7 "Peaceful Uses of Atomic Energy". In Proceedings of 4th International Conference on the Peaceful Uses of Atomic Energy. Geneva, 9/71, Vol. 3, p. 382.

9.0 OTHER TOPICS

The following items cover a spectrum of issues posed to the Task Group.

9.1 PIPING EXPERIENCE DURING EARTHQUAKE

Earthquake loads can potentially affect pipe failure in both fracture and rupture modes. In the ductile fracture failure mode, the earthquake may not contribute greatly to the crack growth due to the limited number of load cycles associated with earthquake motion. However, if the crack has already reached or nearly reached an unstable condition, a severe earthquake motion could conceivably push the crack to a guillotine type of break. It is essential therefore that a proper design safety margin be assigned to prevent the crack from reaching unstable conditions.

Earthquake motion may have a higher potential to induce the rupture mode of pipe failure if piping systems are designed or installed improperly.

A summary of the limited field data on piping experience during earthquakes is listed for discussion in Table 9-1. A detailed survey on piping experience during earthquake is being conducted through the NRC Piping Review Committee Task Group on Seismic Design. More information is available in the Seismic Task Group Report.

From the piping damage record in Table 9-1, two points are noteworthy:

- Not all of the facilities listed are nuclear power plants. Seismic design requirements for these facilities were not as stringent as for modern nuclear power facilities. Consequently, piping systems in these facilities are typically more flexible than modern nuclear piping systems. All the piping systems in the facilities listed survived with litle or no earthquake damage.

- In most cases the earthquake levels were not high enough to induce pipe rupture. Only the Tangshan earthquake resulted in one rupture type of pipe failure, which was caused by support failure.

The observed failure of cracked piping (e.g., El Centro) supports the argument given in Section 9.1 that the leak-before-break concept can only be

Table 9-1 Summary of Observed Seismic Behavior of Piping in Industrial Facilities

Site, Date	Max. Ground Acceleration(g)	Observations
Long Beach, CA 3/10/33	0.25	Five steam plants either operated through the earthquake or were shut down due to loss of load and were back in operation the same day. The five steam units were designed with at most static methods to a 0.2-g level. No piping was damaged.
Kern County, CA 7/21/52	0.26	Oil fired 60 MW steam plant was shut down after the earthquake due to loss of load but was returned to service in a few hours. Piping design based on response spectrum normalized to 0.1 g at ground level, and 0.3 g at the top floor of the buildings. No piping was damaged.
Alaska 1964	--	A power station at an air base had no damaged piping although there were some bent hanger rods. A second power plant in the earthquake zone incurred more damage, but there was no failure of power piping.
San Fernando, CA 1971	>0.25	Valley Power Plant was perhaps designed to a static g-level of 0.2 or 0.25. The plant was tripped off line by action of sudden pressure relay and loss of load. It was back on line inside 2 hours. Other than insulation, the piping was undamaged.
Managua, Nicaragua 1972	0.39	ESSO Refinery. Design for 0.2-g static horizontal load. The facility was shut down for inspection but was operating at full capacity within 24 hours. No damage to piping.

Table 9-1 (Continued)

Site, Date	Max. Ground Acceleration(g)	Observations
Managua, Nicaragua 1972	0.6	ENALUF Power Plant. Earthquake design, if any, not known. No damage to piping.
Imperial Valley, CA 10/15/79	0.5 horiz. 0.66 vert.	El Centro Steam Plant. Earthquake Design based on 0.2-g static horizontal load. No high-temperature or high-pressure piping failed during the earthquake. However, a Victaulic coupling on a straight section of 2" pipe was damaged. Additionally, 3" and 4" water lines failed in straight runs in areas which had been either weld repaired or extensively corroded. Circumferential cracks in these corroded lines, apparently caused by the earthquake, were observed; however, leakage was minimal since the cracks were later found to be essentially closed.
Tangshan, China 7/28/76	Intensity 9	Failures occurred at four loca- along a 60-mile-long crude oil pipeline. The piping system was not designed for any earthquake load, and construction occurred in the early 1900s. Design criteria and design codes were not known. The pipe material's yield strength is 3,500 kg/cm^2, and the ultimate strength is 5,200 kg/cm^2. Pressure range is 60 kg/cm^2 and temperature is 65-70 C. Four separate failures were reported: (1) ring type buckling, (2) four wrinkles on the inside of a bend (3) leakage, (4) rupture.

Table 9-1 (Continued)

Site, Date	Max. Ground Acceleration(g)	Observations
Tangshan, China 7/28/76 (Cont.)		The first three failures occurred at the locations where the pipeline crosses over an active fault. The last failure occurred at the location where the supporting highway bridge was destroyed during the earthquake.

applied to piping systems where crack growth is closely controlled during stable conditions, and where preventive measures are taken against pipe rupture due to causes other than crack growth.

9.2 HEISSDAMPFREAKTOR (HDR) OR SUPERHEATED STEAM REACTOR DOUBLE-ENDED GUILLOTINE BREAK

On November 5, 1983, a double-ended guillotine break occurred at the HDR (Heissdampfreaktor, or Superheated Steam Reactor) test facility in Kahl, West Germany, during system pressurization prior to a blowdown test.* The break took place in the VKL (Versuchskreislauf) piping system that connects the HDR reactor pressure vessel with an external electrically heated boiler, near the joint between a 300 mm-to-80 mm conical reducer and the 300-mm elbow attached to the former superheated steam generator (now used as the VKL system pressurizer). The reducer location, break location, and pipe configuration before and after the break are shown in Figure 9-1.

The break occurred at a system pressure and temperature of 107 bar (1,572 psia) and 315 C (600 F). The RPV was manually isolated about 6 minutes after

* The HDR is a decommissioned nuclear power plant now used strictly as a test facility. The present facility contains no nuclear materials.

the break during which time system pressure fell to 97 bar (1,425 psia). Coolant escaped unimpeded from both ends of the break; because initial system conditions were well above saturation conditions, vaporization already had begun after a pressure drop of 0.5 bar. Damage induced by the break included the following:

- The 80-mm pipe, together with two valves of 450 kg combined weight, whipped two to three times. The force of the pipe whip caused the lower valve and reducer cone to separate as a unit from the vertical pipe section and to be hurled upwards, coming to rest near the containment wall about 5 m from their initial position.
- The blowdown jet escaping from the 300-mm elbow displaced several neighboring pipes of smaller diameter some two to three meters from their original positions.
- The vertical section of the 80-mm pipe bent through a 290° angle, developing a kink at the location shown in Figure 9-1.

The reducer was originally machined in a single piece from a billet of 15 Mo 3 stainless steel. Examination of the broken reducer indicated that it had failed as a result of faulty fabrication. As shown in Figure 9-2, the wall thickness measured after the break was 5.5 mm at the break plane, compared to its design value of 20 mm. Wall thicknesses as thin as 4.4 mm, apparently resulting from plastic deformation during the break process, were measured afterwards.

Approximately 60 distinct flaws were observed on the reducer inner surface, two of which had nearly penetrated the wall. In addition, two lathe grooves were present, about 2 mm apart and 0.1 mm deep. Microscopic examination of the flaw surfaces indicated clear evidence of corrosion.

The results of these observations, together with accompanying theoretical and experimental investigations, led the HDR program staff to the following conclusions:

- The double-ended break resulted from corrosion cracking driven by high local stresses due to faulty fabrication. Crack growth was further influenced by high oxygen concentration (8 ppm) in the system coolant during recent pressurized thermal shock tests. Owing to operating procedures that carefully controlled system heatup and cooldown, as

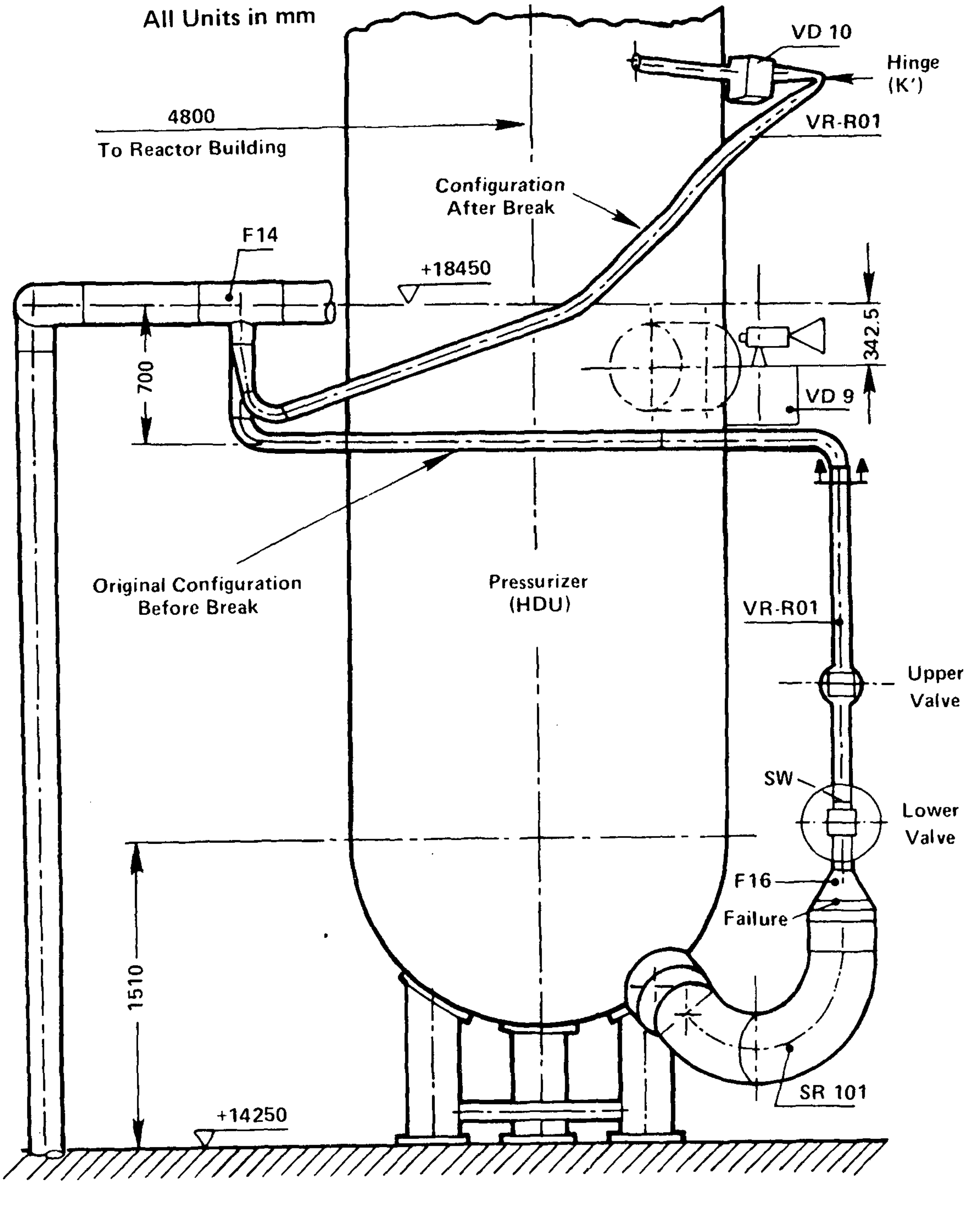

Figure 9-1 Broken Pipe Run in HDR VKL System, Showing Configurations Before and After Break

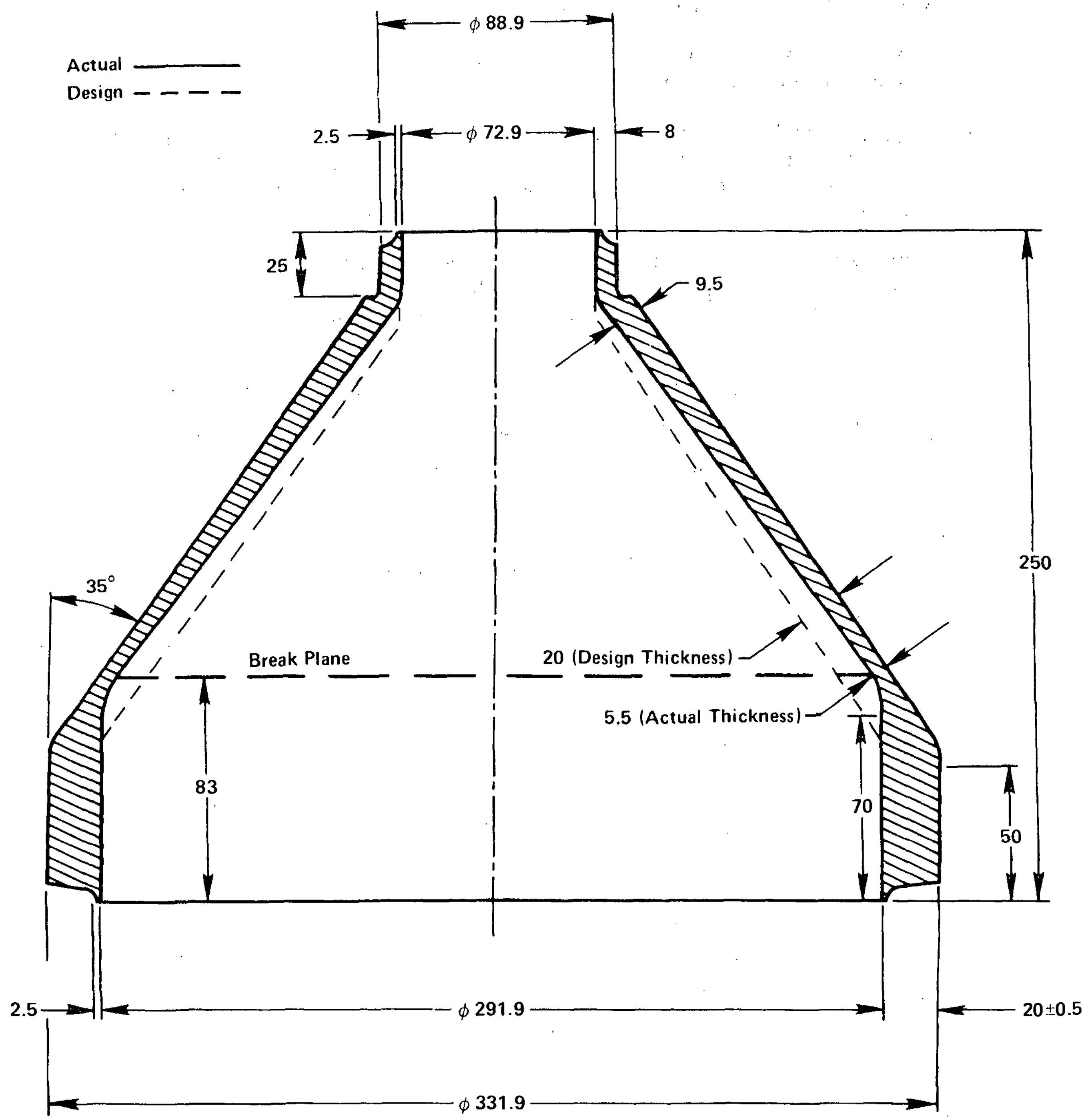

Figure 9-2 HDR Reducer, Showing Design and Actual Wall Thickness

well as the absence of severe operating transients, thermal fatigue was excluded as a cause of failure.

- The ligament remaining after the crack penetrated the wall failed in a primarily ductile manner.

- Fracture mechanics tests conducted in air and under simulated facility operating conditions (i.e., high oxygen content, high temperature water) confirmed that crack growth resulted from corrosion and from excessive stress.

- Autoclave tests indicated for the 15 Mo 3 material that stable crack growth under constant load is possible at stress intensities above 30 MPa $\sqrt{m}$ (28 ksi $\sqrt{in}$) under the given coolant conditions (200 C, 8 ppm O_2 content).

From a review of available measurement data recorded at the time the break occurred, the HDR program staff further concluded that leak before break did not occur. This conclusion was based on the following:

- no deviations in system pressure were observed, other than minor fluctuations characteristic of normal operation.

- no change in the break compartment humidity was observed. It was estimated that a leak rate of about 2 dm^3/hour would have caused about a 10 percent increase in humidity considering the volume of the break compartment.

- no change in ambient temperature was observed.

- a microphone positioned next to the RPV nozzle about 12 m above the break location detected no unusual noise prior to the break.

- acoustic emission measurements on the nozzle nothing unusual prior to the break.

The macroscopic and microscopic flaw surface examinations made by MPA Stuttgart, as the theoretical leak-before-break curves, offer further evidence supporting this conclusion.

9.3 MAJOR PIPE FAILURE PROBABILITIES

Questions have been raised concerning the overall probabilities of major pipe rupture in LWRs. The questions raised were:

- What are the probabilities of major pipe ruptures in various PWRs?
- What is the probability of a major pipe rupture in a BWR?
- How do flawed pipes increase the probability of a major pipe break?
- Why are the above numbers different from those on WASH-1400?

Studies at Lawrence Livermore National Laboratory (LLNL) resulted in specific probabilities of DEGB in reactor coolant loop piping for Westinghouse and Combustion Engineering PWRs. Also, studies are underway with regard to Babcock & Wilcox PWRs and General Electric BWRs. The relevant major primary pipe failure probabilities for Westinghouse and CE plants are given in Tables 3-1 and 3-2 for direct DEGB and Tables 3-3 and 3-4 for indirect DEGB.

Values for B&W PWRs are expected to be similar within a few orders of magnitude.

BWRs pose a different problem because of intergranular stress corrosion cracking (IGSCC). The incidence of IGSCC is far higher than all other failure mechanisms in large piping; however, the toughness of the austenite alloy leads to an anticipation of leak-before-break controlling rather than large break.

The use of nonnuclear pressure vessel failure statistics as a surrogate for nuclear primary piping yields failure values well below 10^{-6} per system-year.

The question regarding flawed pipes presumably relates to fabrication flaws. Extensive work by the British Welding Institute and the Welding Research Council - Pressure Vessel Research Committee confirms that many types of weld defects are relatively innocuous. Since all major systems undergo several levels of nondestructive examination, the probability that a flaw approaching critical size exists is considered very small. Experience has confirmed that the operationally induced flaw is of greater concern than the fabrication flaw.

WASH-1400 piping failure probabilities tend to be higher than experienced and later probabilistic studies indicate this to be true. In fact, the NRC has a study underway at EG&G to examine available data to permit a reevaluation of the original WASH-1400 numbers.

10.0 RECOMMENDED RESEARCH

The approach to demonstrating leak-before-break presented in this report is based on the current state-of-the-art calculational and experimental techniques and is believed to provide a reliable method for evaluating the validity of leak-before-break behavior. Significant experimental and analytic research has been conducted to define the conditions under which leak-before-break is the applicable mode of failure. Much of this research has been conducted in foreign countries, particularly Germany and Japan. The results from these programs support the development of leak-before-break criteria. A review of these programs is presented in Appendix H of NUREG/CR-3142, "The Development of a Plan for the Assessment of Degraded Nuclear Piping by Experimention and Tearing Instability Fracture Mechanics Analysis". Nevertheless, there are several areas where ongoing and additional research can be used to confirm, improve, and expand upon the leak-before-break evaluation method. Especially desirable is a better J estimation procedure that is validated by experiment to eliminate the apparent discrepancies of the currently available procedures as discussed in Section 5.0 and Appendix A of this report. This section identifies the research which is currently underway and recommends additional areas of research that would enhance the leak-before-break assessment procedure.

10.1 FULL-SCALE PIPE FRACTURE EXPERIMENTS

The majority of the relevant pipe fracture experiments conducted to date have been performed on pipes in the range of 2 to 8 in. in diameter with a few tests on 16-inch-diameter pipe. In addition, most of these tests have considered pure bending loads. Only a few tests have been conducted under combined axial and bending load. Experiments on large-diameter pipe and experiments under combined loading conditions should be conducted to confirm and improve on the ductile piping fracture mechanics analysis techniques. Additionally, experiments on welded pipes with cracks located in the base metal, weld metal, and weld heat-affected zone are desirable to demonstrate that the currently used fracture mechanics analyses adequately model realistic field conditions.

Ductile crack extension data from these tests will also provide an important benchmark for J-resistance curve data generated using laboratory size specimens. These benchmark data are essential in determining the most appropriate laboratory specimen test for predicting the fracture behavior of actual piping systems.

Tests of the type described above are currently planned in the NRC Degraded Piping Program and are scheduled to be conducted over the next two years. These efforts are focused on fully ductile fracture modes and not assessing fracture for piping susceptible to fracture in the temperature range where the material is in the brittle-to-ductile transition region. In addition, ENEA in Italy is also planning to run large-diameter pipe tests during the next few years. Other related major pipe fracture research programs are also underway at EPRI, MPA-Stuttgart, and JAERI. Smaller efforts are also underway at Framatome, CEGB, and NUPEC.

10.2 TENSILE AND DUCTILE FRACTURE TOUGHNESS PROPERTIES DATABASE

A comprehensive material properties database for the piping materials commonly used in light water reactor power plants should be developed. This database should include a complete materials charaterization for base and weld materials including chemical compositions, fabrication history, tensile properties, impact properties, and J-resistance curves. Raw data (load displacement curves) should be included for the J-resistance curves to allow evaluation of new or improved J-estimation schemes. Tensile properties should be developed from the elastic range through the maximum load. Work is already being sponsored in this area by the NRC in its Structural Integrity of Water Reactor Pressure Boundary Components program with Materials Engineering Associates, the David W. Taylor Naval Ships Research and Development Center Program and the Degraded Piping Program. The Electric Power Research Institute is also supporting piping material properties data development at Westinghouse. Under the NRC program all the materials property data generated and collected from other programs will be entered into a computerized database management system. Experimental laboratories involved in J-resistance curve testing should pay close attention to developments in the area of improved J-resistance curve specimens. As mentioned above the standard specimen geometries currently in use generally do not provide adequate crack extension for use in ductile fracture mechanics analyses. Thus, efforts should be focused on developing improved standardized specimens that will provide the necessary data at large crack extensions to predict actual pipe behavior.

10.3 PIPING COMPONENT AND COMPLEX PIPE GEOMETRY ANALYSES AND EXPERIMENTS

Ductile piping fracture mechanics analyses and experiments conducted to date are limited primarily to straight piping sections. It is recommended that fracture experiments be conducted on complex piping geometries including cracks located in nozzles and near elbow and other piping components. These experiments would provide a benchmark for determining the adequacy of current fracture analyses when applied to these more realistic situations. In addition, testing of entire piping systems would provide a good benchmark for evaluating the entire leak-before-break evaluation procedure that relies on loads taken from design stress reports, piping system compliances generally determined using piping design computer codes, and the fracture mechanics analysis. A determination of the overall margins of safety would require testing of a prototypical piping system for which all the above analyses could be made and the analytic and experimental results compared. To the Task Group's knowledge no organization is currently supporting such testing.

10.4 SIMULATED SEISMIC LOAD TESTS

The ductile fracture mechanics analysis and experimental J-resistance curve techniques discussed in the report assume that loads are applied in a monotonically increasing fashion. In reality, under seismic loading conditions fully reversed cyclic loading could be anticipated. To date little work has been performed to evaluate the load history effects on ductile fracture toughness properties. NRC is planning to investigate this area in small specimen tests to be conducted at the David W. Taylor Naval Ship Research and Development Center. Additionally, tests of flawed piping systems subjected to simulated seismic loading would provide for determining the adequacy of the current monotonic, quasi-static loading assumptions and developing improved analytical techniques, if necessary. These types of tests may couple well with simulated seismic testing of components and unflawed piping systems currently being planned by EPRI and NRC. Some foreign research is also being planned in this area. Specifically, dynamic load testing of flawed pipes is being planned by MPA to be conducted at the decomissioned HDR plant in the Federal Republic of Germany.

10.5 LEAK RATE TESTING AND LEAK RATE DETECTION

Knowledge of the leak rates associated with various postulated through-wall crack lengths and confidence in the ability to detect leakage in a timely manner is a critical element of the leak-before-break concept suggested in this report as a basis for eliminating the postulated double-ended guillotine break. Additional data are necessary to further validate and improve existing leak rate prediction analyses. This conclusion was reached by the attendees at a special CSNI meeting on leak before break in light water reactor piping systems held in Monterey, California, in September 1983. The meeting attendees concluded that additional leak rate testing should be performed to provide greater confidence in existing leak rate calculations.

Improved leak rate detection systems should be pursued to provide additional confidence in the leak-before-break concept. Of particular interest would be investigation and improvement of local leak detection systems such as acoustic emission monitors or moisture-sensitive tapes since these techniques may be important for establishing the validity of leak before break at specific piping system locations.

10.6 HIGH ENERGY TESTING

High energy testing refers to the testing of pipe containing water under light water reactor pressure and temperature conditions. These types of tests involve the release of tremendous amounts of energy and are difficult and expensive to conduct. The value of performing these tests would be in identifying the need for replacement criteria for the double-ended break and in defining such criteria. To perform these tests the loading structure would have to be designed to accommodate a variety of realistic piping system compliances. This would allow determination of crack opening areas, crack opening times, and jet impingement and blowdown reaction loads under conditions that represent realistic operating conditions. The results of these tests would be useful in reevaluating the current methods for defining post-break dynamic loads and other post-break phenomona in systems for which leak before break will not be applied at this time.

10.7 WATER HAMMER TESTING

A limited number of simulated water hammer tests are currently planned under the NRC Degraded Piping Program. These tests will be important for defining the loads associated with water hammer events and observing the ductile fracture response of piping subject to the dynamic loads associated with water hammer. Results of these experiments should give some insight to the limitations of applying leak before break to piping systems subject to water hammer.

The above recommended research is important not only with respect to the leak-before-break concept as applied to the elimination of the double-ended break but also to other flawed pipe evaluations. Note that because of declining research budgets these research efforts will only be accomplished through careful planning and coordination of research efforts and possibly pooling of funds to support some of the more expensive tasks. Such cooperative efforts are currently being pursued through the development of an International Piping Integrity Research Group (IPIRG).

11.0 SUMMARY: CONCLUSIONS AND RECOMMENDATIONS

The Pipe Break Task Group has developed the following conclusions and recommendations. They are listed by specific sections of the report.

11.1 CONCLUSIONS

In Section 5.0 and Appendix A of this report, the Task Group describes a comparison of analytical results calculated by various J-estimation procedures with the results of actual experiments. Based on this comparison, the Task Group concludes that there can be significant differences between analysis and experiment. The discrepancies can be conservative or nonconservative depending on the estimation procedure used (see Table A-3 and Figures A-7, A-8, A-9 and A-10 of Appendix A). In general, the EPRI-estimation scheme was found to be conservative while the NUREG/CR-3464 procedure was nonconservative in predicting the value of J at crack initiation. The NRC staff's modification of the NUREG procedure to include the effect of material strain hardening resulted in a better fit to the experimental data, but on the average was still somewhat nonconservative.

These results indicate that there can be significant computational differences. The Task Group believes that the computational uncertainty is appropriately accounted for by the margins specified in Section 5.2 of the main text. However, the analyst should take steps to ensure that significant nonconservative predictions are not made.

6.3 Any detailed value-impact assessment of changes in pipe break criteria requires knowledge beforehand of the specific changes themselves. Nevertheless, a review of assessments already completed for specific pipe break issues implies the following general conclusions for elimination of DEGB as a design basis for PWR reactor coolant loop piping:

- Elimination of pipe whip restraints would only negligibly increase public and occupational radiation exposure (ORE) resulting from pipe break accidents.

- Elimination of pipe whip restraints would improve access to pipe welds for ISI, and thereby significantly reduce ORE during inspection. Improved access would also reduce ORE during normal plant maintenance, although to a lesser degree. This benefit would apply both to operating plants and to plants under construction.

- For operating plants not already having pipe whip restraints, eliminating DEGB -- and thus the need to install restraints -- would reduce ORE during installation. For the sixteen A-2 plants, installation and maintenance of these restraints would avoid the occurrence of ORE by about 11,000 man-rem compared to a small increase in public risk and accident avoided occupational exposure of less than 5 man-rem.

11.2 RECOMMENDATIONS

3.5.3 Revise Standard Review Plan 3.6.2 (MEB 3-1) to incorporate a proposed change to eliminate the requirement to postulate arbitrary intermediate breaks in high energy lines. Environmental qualification of safety-related equipment in the vicinity of these lines should be reviewed on a case-by-case basis.

4.1 Plant-specific exemptions to GDC-4 should be granted for LWR applications and licensees that provide justification* for such requests both on a technical and safety benefit basis. Such exemptions should relate to the requirement to postulate pipe breaks up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. Further, the scope of the exemptions only should be applicable to the measures required for protection against the dynamic effects (e.g., pipe whip, jet impingement) of postulated pipe ruptures; it should not pertain at this time to the definition of a loss-of-coolant accident (LOCA) nor its relationship to the regulations addressing design requirements for ECCS (10 CFR Part 50.46), containment (GDC-16 and -50), other engineered safety features. In parallel with the granting of exemptions to GDC-4, the staff should expedite rulemaking to address this issue.

5.9.4 Based on the discussion in Section 5.9.1 and 5.9.2 and Appendix A the Task Group has the following recommendation regarding application of ductile piping fracture mechanics to leak-before-break evaluations.

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*NRC Generic Letter 84-04, dated February 1, 1984.

- As indicated in Section 5.9.1 there are certain limitations that currently preclude generic use of limit load analysis to evaluate leak-before-break conditions for eliminating pipe restraints. However, the task group believes that limit-load analysis can be used to demonstrate acceptable leak-before-break margins for the application provided the limit moment determined from Eq. (A-19) in Appendix A is greater than the applied (normal operation plus SSE) moment at any location in the pipe run by a factor of at least three. Limit moment should be determined from Eq. (A-19) in Appendix A when the flow stress is determined from ASME Code minimum properties. Data obtained from future tests (see Section 10.0) may provide information that would allow less restrictive use of limit-load analysis for justifying elimination of pipe restraints.

- When crack extension is predicted to occur, stability analysis should be performed (see Section 3.4) to determine if adequate margins against crack instability are maintained. Stability computations should include crack extension characteristics of the materials as defined by appropriate J-R curve data.

- The stress-strain curves should be obtained over the range from the proportional limit to maximum load.

- Ideally, the materials tests should be conducted using archival material for the pipe being evaluated. If archival material is not available, tests should be conducted using specimens from three heats of material having the same material specification. Test material should include base and weld metals.

- Three J-R curves and three stress-strain curves should be generated for each previously defined material. The tests should be conducted at temperatures near the upper range of normal plant operation (e.g., 550 F). Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. One J-R curve and one stress-strain curve for one base metal and weld metal are considered adequate to determine temperature dependence.

- The piping materials toughness (J-R curves) and tensile (stress-strain curves) properties should be determined at temperatures near the upper range of normal plant operation. The test data should demonstrate ductile behavior at these temperatures.

- The J-R curves should be obtained using specimens whose thickness is equal or greater than that of the pipe wall. The specimen should be large enough to provide crack extension up to an amount consistent with J/T condition determined by analysis for the application. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques may be used if appropriate as described in Section A2.4 of Appendix A.

APPENDIX A

FRACTURE MECHANICS ANALYSIS

APPENDIX A - FRACTURE MECHANICS ANALYSIS

A1.0 INTRODUCTION

Fracture mechanics analyses have been used in the nuclear industry over the years to assess the integrity of a variety of components with known or postulated defects that can be modeled as cracks. Typically these analyses are performed to determine if known or postulated defects will compromise component availability and reliability during subsequent in-service operation.

More recently, fracture mechanics analyses have been and are being used to demonstrate that ductile nuclear piping systems have sufficient fracture resistance to preclude the necessity for postulating pipe breaks. The objective of these analyses is to determine if piping systems subjected to large postulated accident loads can tolerate, with acceptable margin against failure, the presence of relatively large postulated cracks. To allow implementation of fracture mechanics methods in lieu of postulating pipe breaks, guidelines have been developed (see Section 5.2 of main text) to define acceptable procedures and practices for performing the fracture mechanics analysis.

Briefly, these guidelines discuss the currently available ductile fracture mechanics analysis algorithms and the need for appropriate associated materials properties; the use of a throughwall crack that can be detected during normal operation by in-plant leakage monitoring systems; the use of postulated accident loads; and recommended margins on postulated load or flaw size to ensure adequate resistance to fracture.

The purpose of this appendix is to discuss the ductile fracture mechanics methods that are available to perform leak-before-break analyses consistent with the guidelines presented in Section 5.2 of the main text. This discussion includes various computational techniques, benchmark comparisons with currently available pipe experimental results, and consideration of appropriate materials data for use with the analysis methods.

It is the intent of this appendix to compare various ductile fracture mechanics analytical procedures with one another and to relate their results to available experimental data. Detailed descriptions of the procedures are available in the literature and are not discussed in depth here. Rather, it is the intent of the Task Group to assess the state of the art that these methods represent.

Ductile fracture mechanics (FM) methods employ analytical techniques ranging from elaborate finite-element models (FEM) to various FM estimation procedures to simple limit-load analyses. FEM analyses are expensive and time consuming to perform and the purpose of the simple models is to facilitate the performance of FM analyses in a timely and relatively inexpensive manner.

Although all FM methods are based (at least to some extent) on theory, it is necessary to include in them certain idealizing assumptions related to crack shapes, consistent geometry and crack behavior if the crack initiates and grows as a result of increased loads. Also under most circumstances, it is necessary to obtain materials property data from other than the component being evaluated.

In real life, however, actual flaws can have complex shapes, the component being evaluated may deform under high loads particularly in the vicinity of the flaw (e.g., a pipe may ovalize and its wall may become thinner near the flaw) and a growing crack may develop shear lips. These reasons plus the inherent variability of material properties from specimen to specimen lead to the conclusion that perfect correspondence between analytical and experimental results should not be expected. On the other hand, to be useful at all, analytical methods should be able to predict results within an acceptable uncertainty band which can then be accounted for by appropriate margins.

A2.0 DESCRIPTION OF ANALYTICAL METHODOLOGY

A2.1 DUCTILE FRACTURE MECHANICS EVALUATION PROCEDURES

Several methods currently are available to analyze and evaluate leak-before-break conditions in ductile piping with postulated throughwall flaws. These methods include but are not limited to limit load (net section collapse) analysis, the J integral/tearing modulus (J/T) approach, the R-6 approach and its derivative the failure assessment diagram (FAD), and crack tip opening angle (CTOA).

Although each of the previously listed assessment methods can be successfully applied in many cases, the NRC leak-before-break criteria are largely based on the J/T approach (see Section 5.0 in the main text), which is the focus of attention in this appendix. The J/T approach has been selected because it is a general procedure that incorporates a rational crack tip parameter, can discriminate between materials of different toughness and tensile properties, and can incorporate various boundary conditions (e.g., load vs displacement control) and pipe system characteristics (e.g., system configuration and support characteristics). The R-6 and FAD methods are similar in nature to the J/T approach and use several of the same variables in their application; consequently, they are not discussed in detail here. The interested reader may refer to References A.1 and A.2. The CTOA method typically is not used for nuclear applications in the United States. Limit load analysis often provides an adequate means to determine if leak-before-break conditions are satisfied. However, limit-load analyses may not adequately represent crack extension resistance over the range of possible anticipated applications (see Section 5.9.1 of the main text). Ongoing experimental work (see Section 10.1 of the main text) will provide additional data concerning the general application of limit load analysis for leak-before-break evaluations.

A2.1.1 Illustration of J/T Approach

Two important aspects should be considered in general when evaluating crack extension for leak-before-break analyses, namely, initiation or first extension of the existing flaw and stability of a growing flaw subsequent to initiation. The material value of J associated with initiation of additional crack extension is denoted as J_{IC}. If the applied value of J is less than J_{IC}, crack initiation or significant growth will not occur and stability of the existing crack is ensured automatically. When extension of the existing crack is predicted, the crack extension must be evaluated to determine if it occurs in a stable manner, or if the crack will grow unstably and result in a predicted full break.

A convenient means now commonly used to define the margin against instability involves plotting J as a function of T for the applied and material resistance values. This J-T diagram is shown schematically in Figure A-1. Here, the material curve is developed from a J-R curve illustrated in Figure A-2. Three J/T lines representing elastic-plastic displacement control loading (curve A), elastic-plastic load control loading (Curve B), and fully plastic displacement control loading (Curve C) are presented in Figure A-1. The applied tearing modulus curve is developed using formulas that are presented later in Section A.2.3 of this appendix.

If the applied load, crack length, and system parameters are such that the applied curve intersects the material curve, then crack instability is predicted. In Figure A-1, points I define the instability points for the respective assumed conditions.

The J-T diagram is used within two bounding limits. If the applied J is below J_{IC}, then crack stability is automatically assured because crack growth is not implied. The upper bound limit for J-controlled growth is illustrated as point L on the J-T diagram. Beyond point L, certain assumptions in the J-T formulation may not be satisfied (see Section A2.2). In this instance, the tearing stability methods can be applied approximately and the analyst must use caution in interpreting results (see Section A2.4).

Each of the methods used to define the applied J/T line can be applied successfully for leak-before-break applications. From a practical standpoint predictive methods based on elastic plastic displacement control conditions are not available for generalized analysis of real, complex piping systems. Consequently, J/T analyses are typically based on either elastic-plastic load control or fully plastic displacement control analyses.

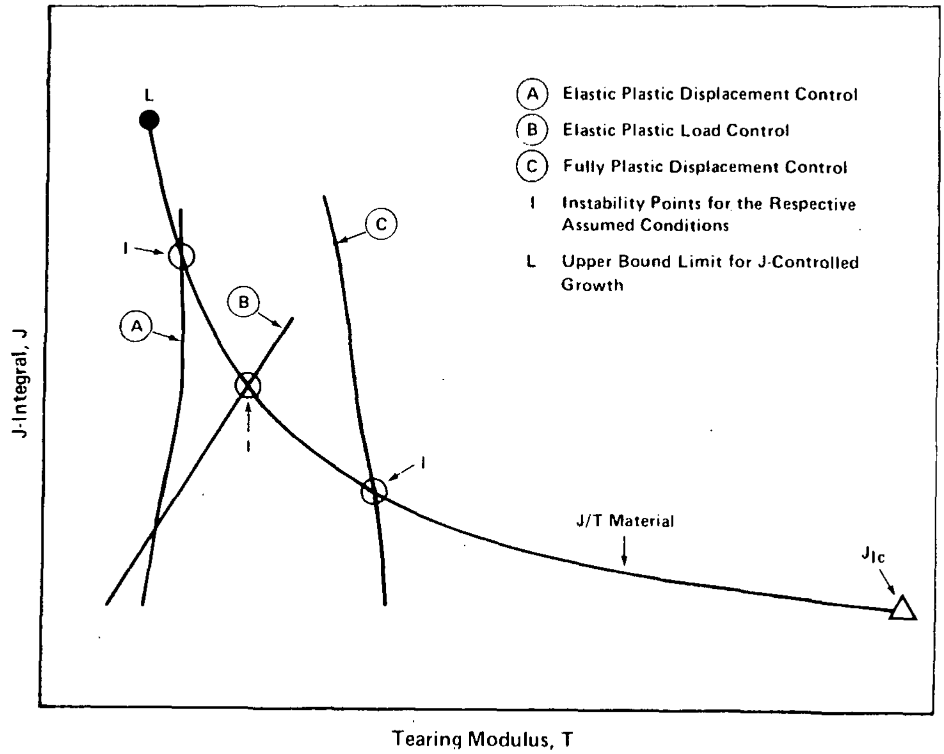

Figure A-1. J/T Plot

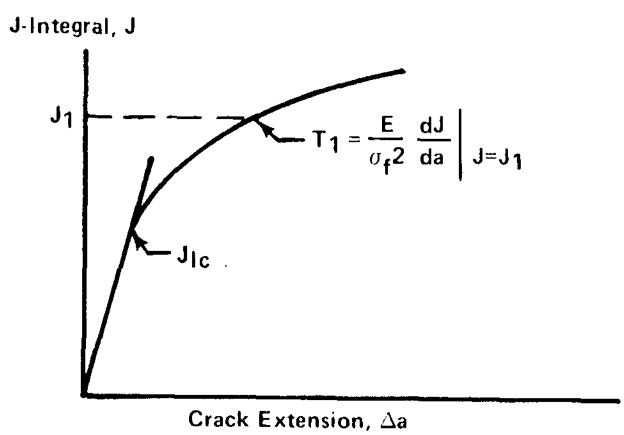

Figure A-2. Material Resistance Curve (J-R Curve)

The analytical basis for the J/T approach, computational schemes for J and T and the materials data required for application of the method are described in Sections A2.2, A2.3 and A2.4, respectively.

A2.2 ANALYTICAL BASES FOR J/T METHODOLOGY

There are three basic considerations in the tearing modulus (J/T) approach. The first consideration requires the equilibrium between the parameter describing the potential to extend an existing crack, typically written as J in the literature, and the material resistance to crack extension, J_m. The equilibrum condition is expressed mathematically as

$$J = J_m\ (\Delta a) \tag{A-1}$$

J is a measure of the elastic-plastic stress-strain field around the crack tip for any specified crack geometry and loading. Since its introduction[A.3] in 1968, J expressions have been developed for various flaw geometries and loadings.[A.4, A.5, A.6, A.7] The J formula for a pipe flaw geometry will be discussed later in this section. Here, it is sufficient to note that J depends on the geometry of the flawed component, flaw shape, orientation, and loading type (tension, bending, etc.). J also depends upon the material stress-strain relationship as it dictates the extent of plasticity in the vicinity of the crack tip.

The material resistance to crack extension, typically referred to as the J-R curve and illustrated in Figure A-2, is considered to be a material property for a specific heat of material, temperature, and a crack-related condition called plane strain. In reality, however, J-R curves are also often found to depend upon parameters such as type of loading (tension or bending), crack geometry, and component thickness[A.5, A.8, A.9, A.10]. In the J-R curve shown in Figure A-2, J_{Ic} refers to the onset of extension of the existing crack. Where the plane strain conditions are satisfied, initiation J is denoted by J_{Ic}. Plane strain crack condition generally provides a lower bound behavior for material resistance to stable crack growth.

The second consideration in the tearing modulus approach is that proportional loading of the crack tip field must be satisfied during crack growth.(A.11) The condition for the proportional loading (J-controlled growth) is expressed as

$$\omega = \frac{dJ}{da} \cdot \frac{b}{J} > > 1, \qquad \text{(A-2)}$$

where b is the remaining ligament, and the term on the left side of inequality generally is denoted in the literature by the greek symbol ω. Details on J-controlled growth can be found in References A.11 and A.12; it is sufficient to note here that ω greater than 10 would generally satisfy J-controlled growth requirements and ensure that the J/T theory is applicable. This requirement must at least be satisfied by the J-resistance curve. Generally, only small amounts of crack growth are allowed under the strict requirements of J-controlled growth.

The third aspect of the J/T approach concerns stability of a growing crack. While Eq. (A-1) provides a means for inferring crack growth from the J-resistance curve, it does not define stable crack growth. Crack growth stability is evaluated by comparing the applied tearing modulus against its material counterpart.(A.13) For stable crack growth this is expressed as

$$\frac{dJ}{da} \cdot \frac{E}{\sigma_f 2} < \frac{dJ_m}{da} \cdot \frac{E}{\sigma_f 2} \qquad \text{(A-3)}$$

where the term on the left side of the inequality is the applied tearing modulus, T, and the right side term is the material counterpart, T_m. The term dJ_m/da is the slope of the J-R curve (at J greater than J_{IC}) illustrated in Figure A-2. The flow stress, σ_f, usually defined as the average of yield and ultimate strengths, should be determined experimentally. E is the elastic modulus. $E/\sigma_f 2$ is a normalizing term which was originally introduced with dJ_m/da to eliminate the temperature dependence of the resistance curve; however, subsequent work has shown that T_m is not independent of temperature effects.

Eq. (A-3) can be more simply written as

$$T < T_m \tag{A-4}$$

for stability of crack growth.

The above discussion on crack stability is an extension of the graphical approach used in linear elastic fracture mechanics (LEFM) methods. In LEFM methods, crack instability is normally evaluated under load-controlled conditions. The tearing modulus concept extended this idea to more realistic conditions such as a displacement-controlled loading for a compliant system. The displacement-controlled loading is one where displacements (rotations) at certain reference locations are held fixed while examining crack growth stability. Such loading allows system characteristics to be readily incorporated into the tearing instability analysis.

A2.3 J AND T COMPUTATIONS

A2.3.1 Computation of J

Elastic-Plastic J-Integral Estimate. The computation of J for the throughwall flaw geometry illustrated in Figure A-3 follows the method described in References A.6 and A.14, where the J integral is separated into elastic and plastic components, as follows:

$$J = J_e + J_p \tag{A-5}$$

where

J_e is the plasticity adjusted elastic contribution to J

J_p is the plastic contribution to J.

The elastic portion of J is directly related to the elastic stress intensity factor, K_I, by the relationship

$$J_e = K_I^2/E \tag{A-6}$$

Elastic K_I solutions are available from References A.15, A.16, A.17, and A.18.

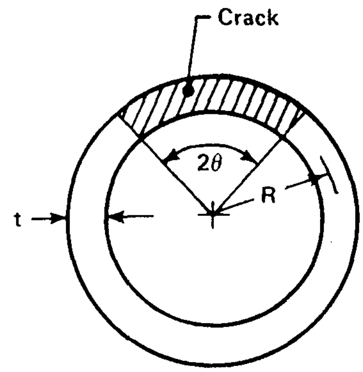

Figure A-3. Pipe Cross Section Containing a Through Crack

Several methods are available to calculate the plastic component of J, J_p. These include direct numerical procedures and various closed form J estimation procedures. The following briefly describes some of the readily available J_p estimation procedures. The first, known as the EPRI estimation scheme(A.6) is expressed for bending loads as

$$J_p = \alpha\ \sigma_0 \varepsilon_0 c h_1 (M/M_0)^{n+1} \tag{A-7}$$

where

σ_0 and ε_0 are the reference yield stress and strain and α and n are material constants determined from the material stress strain curve fit to a Ramberg-Osgood curve*

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* $\varepsilon/\varepsilon_0 = \sigma/\sigma_0 + \alpha(\sigma/\sigma_0)^n$

2c is the remaining circumferential ligament of the cracked portion of the pipe

h_1 is a function which accounts for relative crack and component size, and material work hardening

M_0 is the moment required to develop an average stress of magnitude σ_0 in the cracked section

M is the applied moment.

NUREG/CR-3464 (Reference A.19) describes an alternate procedure for determining the applied J in which J is developed from the kink angle, ϕ, or total rotation of the pipe as illustrated in Figure A-4.

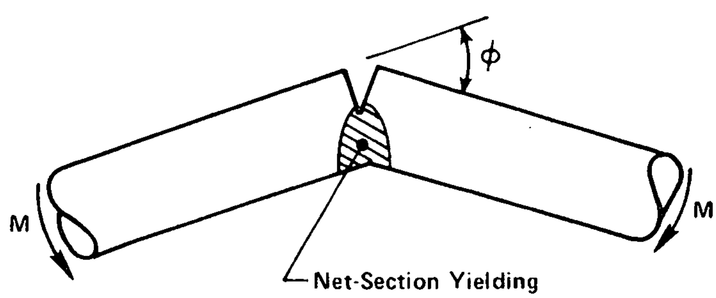

Figure A-4. A Pipe Containing a Through Crack Under Bending

Eq. (13), page 157 of the NUREG document is:

$$\phi = \frac{\sigma_b}{E} I_b(\theta) + \frac{\sigma_t}{E} I_t(\theta) \tag{A-8}$$

where:

ϕ = kink angle
σ_b and σ_t = applied bending and axial stresses respectively
$I_b(\theta)$ and $I_t(\theta)$ = compliance functions defined in the NUREG
2θ = total effective crack angle (Figure A-3) that includes a plastic zone size correction (see discussion, page 88, of NUREG/CR-3464)

Following the practice used in the NUREG document, the kink angle equation is normalized by

$$\bar{\phi} = \frac{E\phi}{\sigma_f}, \quad S = \frac{\sigma}{\sigma_f}, \quad \bar{\varepsilon} = \frac{E\varepsilon}{\sigma_f}$$

where $\varepsilon = \frac{\sigma}{E}$.

The NUREG equation then becomes:

$$\bar{\phi} = \bar{\varepsilon}_b \; I_b(\theta) + \bar{\varepsilon}_t \; I_t(\theta) \tag{A-9}$$

As discussed in Section 5.0 of this report, the NRC staff modified the analytical procedures of NUREG/CR-3464 to include the effects of strain-hardening of piping materials. Otherwise the staff's J analysis procedure is the same as in the NUREG document.

Rewriting the Ramberg-Osgood equation using $\sigma_o = E\ \varepsilon_o$, it becomes:

$$\varepsilon = \frac{\sigma}{E} + \alpha\left(\frac{\sigma_f}{E}\right)\left(\frac{\sigma_f}{\sigma_o}\right)^{n-1}\left(\frac{\sigma}{\sigma_f}\right)^n$$

or in normalized form:

$$\bar{\varepsilon} = S + \alpha' S^n \qquad \text{(A-10)}$$

where the normalizing factors are given above and $\alpha' = \alpha\left(\frac{\sigma_f}{\sigma_o}\right)^{n-1}$.

Substituting $\bar{\varepsilon}_b$ and $\bar{\varepsilon}_t$ in accordance with Eq.(A-10) into Eq. (A-9), the NRC staff's normalized kink angle becomes:

$$\bar{\phi} = [S_b + \alpha' S_b{}^n] I_b(\theta) + [S_t + \alpha' S_t{}^n] I_t(\theta) \qquad \text{(A-11)}$$

In the calculations performed for the comparisons discussed later in this appendix, the axial stress, σ_t, is zero and hence the last term in Eq. (A-11) disappears. The NRC staff recognizes that, when the axial stress contributes significantly to the strain, alternate procedures may be more appropriate for relating the kink angle to combined bending plus axial stresses. One approach being considered in lieu of Eq. (A-11) is:

$$\bar{\phi} = [S_b I_b(\theta) + S_t I_t(\theta)][1 + \alpha'(S_b + S_t)^{n-1}]$$

It is expected that future pipe test results will lead to an appropriate estimation procedure.

<u>Fully Plastic J-Integral Estimate</u>. The formula for the applied J is derived using limit moment conditions(A.17) and is

$$J = \sigma_f R F \phi \qquad \text{(A-12)}$$

where

$$F = \sin\left(\frac{\theta}{2}\right) + \cos\theta \quad . \qquad \text{(A-13)}$$

In the above, ϕ is the total bending or kink angle of the crack section (see Figure A-4) and is the portion of pipe bending associated with the presence of the crack.

The value of J can be calculated from the knowledge of the crack length, pipe radius, flow stress, and kink angle. Unfortunately, the kink angle, ϕ, is not easy to calculate because of the plastic assumption. This complication can be alleviated by assuming a value of the kink angle; Reference A.13 arbitrarily assumed one degree. Theoretically, the kink angle should depend upon the crack length, pipe size, and applied moment (limit moment, in this case).(A.18) However, as shown in References A.4, A.5, and A.19, the J-T plot (Curve C) can be obtained without knowing ϕ.

A2.3.2 Applied Tearing Modulus

Elastic-Plastic Load Control and Displacement Control. The applied tearing modulus for elastic-plastic load control conditions can be expressed as

$$T = \frac{dJ}{da} \frac{E}{\sigma_f^2} \quad \text{(load control)} \tag{A-14a}$$

where J typically is calculated from the relationship in Eq. (A-5).

The expression for $T_{applied}$ for elastic-plastic, displacement control can be expressed for a pipe subjected to pure bending as(A.8)

$$\left(\frac{dJ_p}{da}\right)_{\delta_T} = 4t(\beta P)^2 \left[\frac{(2C_S + C_e)}{1 + (2C_S + C_e) \cdot \left(\frac{\partial P}{\partial \delta_p}\right)_\theta}\right] + \frac{2\gamma}{R} Jp \tag{A-14b}$$

where

a = $R\theta/2$

δ_T = total displacement of the pipe

δ_p = plastic portion of pipe displacement

β = $-h'(\theta)/R \cdot t \cdot h(\theta)$

$h(\theta)$ = $[\cos(\theta/4) - \frac{1}{2} \sin(\theta/2)]$

γ = $h''(\theta)/h'(\theta)$

$2P$ = total load

C_S = spring compliance*
C_e = elastic compliance of uncracked pipe

Fully Plastic Displacement Control Analysis. The applied tearing modulus for a throughwall circumferentially cracked pipe with constant end displacement can be estimated by(A.7)

$$T = C_1 \; (L/R) + C_2 J \tag{A-15}$$

where

$$C_1 = 2F^2/\pi \tag{A-16}$$

$$C_2 = [\cos(\theta/2) - 2\sin\theta] \; E/2FR\sigma_f{}^2 \tag{A-17}$$

F is defined in Eq. (A-13).

The applied tearing modulus in Eq. (A-15) consists of two terms. The first term in the sum contains the pipe length or compliance, whereas the second term contains the applied J. The length term can be expressed in terms of piping stiffness as

$$L = \frac{EI}{k} \quad . \tag{A-18}$$

where I is the area moment of inertia and k is the piping stiffness. The relationships indicated in Eqs. (A-15) through (A-18) show that the L/R term represents the elastic piping stiffness and does not involve the through crack geometry or J level.

* The term C_S, spring compliance, is included here since past compliant pipe instability experiments used springs to simulate long pipe lengths. For plant applications, the system compliance is accounted for in the term C_e and $C_S = 0$.

Piping systems are generally three-dimensional involving complex component geometries. Generally, these complexities dictate that the piping stiffness be determined using a piping computer code. The stiffness for the pipe system specific configuration is expressed in terms of an effective pipe length, L_{eff}, using relationships similar to Eq. (A-18). Because piping system stiffness is a function of location, the L_{eff} used for the applied tearing modulus calculation must be determined for the location where a throughwall crack is postulated. A description of the L_{eff} calculation for single and multiple cracked pipe system is presented in Refs. A.19 and A.23.

Net Section Plastic Collapse. The limit moment corresponding to fully plastic conditions can be determined from Ref. A.24 and is expressed as

$$M_f = 4\sigma_f R^2 t \left(\cos\gamma - \tfrac{1}{2}\sin\theta\right) \tag{A-19}$$

where

R is the mean pipe radius

t is the pipe wall thickness

θ = half crack angle

σ_f = 1/2 (yield + ultimate strength)

$$\gamma = \frac{\theta}{2} + \frac{\pi}{2}\,\frac{\text{Axial Load}}{2\pi R\sigma_f t}$$

In deriving the limit moment, it was assumed that the material stress-strain curve can be modeled as elastic-perfectly plastic having flow stress, σ_f, as the limiting value of stress. See Section 5.9.1 of the main text for a further discussion of limit load analyses.

A2.4 MATERIAL PROPERTIES

The ductile piping fracture mechanics analysis techniques that are applied in the leak-before-break assessment are strongly dependent on the material tensile properties and resistance to ductile crack extension. These material properties must be carefully obtained to ensure their applicability to the piping materials and operating environments of interest. Furthermore, they must be utilized in a manner consistent with the assumptions made in developing the fracture mechanics analysis techniques to ensure proper results. The following subsections provide guidance for assuring the applicability of material properties data and for developing appropriate tensile and ductile fracture toughness properties for use in the fracture mechanics analyses.

A2.4.1 Assuring Applicability of Material Properties Data

Care must be taken to ensure that the materials tested and the conditions under which they are tested are representative of the materials and operating environment of the piping system being evaluated. Assurance that the test materials used are representative of the actual piping system materials should be provided as follows. A review of available design, fabrication, and quality assurance records for the piping system of interest should be performed to characterize the material and fabrication procedures used in constructing the piping system. Where possible, information should be presented on the chemical compositions of the base and weld materials, pipe fabrication procedures, welding procedures, tensile and impact properties, and other pertinent information. Ideally, the material properties will be determined using archival material of the same heat number. When an archival heat of material is not available, at least three heats of material having the same material specification and thermal and fabrication histories should be tested. These heats of material should be selected or fabricated so as to match as closely as possible the chemical composition, fabrication history, and tensile and impact properties of the piping system materials being evaluated.

The range of relevant operating temperatures and any other appropriate environmental parameters should also be considered in developing the material properties data. With regard to temperature, a review should be performed to define the range of operating temperatures associated with normal operating and accident conditions or other operating conditions where large pipe rupture could have adverse effects on safety. Existing data indicate that the resistance to ductile crack extension can increase or decrease with increasing temperature depending on the type of material. Therefore, the material properties should be determined at a temperature near the upper end of the operating temperature and the lowest temperature of concern as defined by the review discussed above. The leak-before-break fracture mechanics methodology described in this report is limited in application to piping systems where material will not exhibit cleavage type fracture under the applicable range of normal and postulated accident conditions where pipe rupture could have significant adverse consequences for the system being evaluated. In addition to temperature, any other environmental conditions that affect the tensile or ductile fracture toughness properties should be considered. For example, for cast stainless steel materials the effects of thermal aging on the tensile properties and the ductile fracture toughness should be taken into account by subjecting the test material to a degree of aging equivalent to that anticipated over the operating life of the piping system.

A2.4.2 Tensile Properties

The ductile fracture mechanics analysis techniques used in the leak-before-break assessment can be sensitive to the assumed uniaxial stress-strain relationship. For closed form estimation schemes, the uniaxial true stress-true strain curve is generally assumed to follow a power-law hardening relationship. The benchmark calculations presented in Section A3.0 indicate that the power-law curve should be made to fit the material stress-strain data between yield and 10-percent strain. This appears to be appropriate from limited sensitivity studies. It is recognized that the range of the stress-strain curve that must be fit to ensure proper results will vary with pipe and crack geometries. However, the experiments that the calculations were benchmarked against are believed to be sufficiently representative of the class of problems of interest (i.e., leakage size cracks at or less than limit load conditions) in leak-before-break evaluations to suggest that the stress-strain data be fit in the low strain region to provide the best results.

To provide adequate data to support the leak-before-break fracture mechanics analysis, at least two stress-strain curves should be developed for each of a minimum of three heats of materials having the same material specifications and thermal and fabrication histories as the in-service piping material. If the stress-strain data are being developed from an archival heat of material, a minimum of three stress-strain curves will be sufficient. These stress-strain curves should be developed at the highest temperature of concern. In addition, at least one stress-strain curve for one base metal and one weld metal should be generated at a lower temperature, as described in Section A2.4.1, to provide information on temperature dependence of the stress-strain properties. Although the range of the stress-strain curve believed to be of greatest interest is the low strain range, it is suggested that the stress-strain curve be developed over its entire range from elastic response to maximum load. These tests should be conducted at conventional strain rates ($\sim 10^{-4}$ sec^{-1}). Higher strain rates are not considered necessary since previous studies show that the tensile and ductile fracture toughness properties have improved resistance to ductile fracture at elevated strain rates.(A.22)

A2.4.3 Ductile Fracture Toughness Data

Material resistance to ductile crack extension should be based on a reasonable lower-bound estimate of the material J-resistance curve. As indicated in Section 2.4.1 of the main text, the lower-bound material fracture resistance should be obtained from either archival material of the specific heat of the piping material under evaluation, or from at least three heats of material having the same material specification and thermal and fabrication histories as the actual in-service piping material. To account for heat-to-heat and test-to-test variability, at least two J-resistance curves should be generated for each heat of material tested, except in the case where only one archival heat of material is tested, in which case a minimum of three J-resistance curves should be generated. These tests should be conducted at the upper and lower temperatures of concern. In addition, at least one J-resistance curve for one base metal and one weld metal should be generated at a lower temperature, as defined in Section A2.4.1, to provide information on the temperature dependence of the ductile fracture toughness properties.

Regarding specimen geometry, existing data indicate that fracture toughness specimens having approximately the same thickness as the pipe wall and without sidegrooves tend to model actual pipe behavior most accurately. Thus, fracture specimens without sidegrooves, and having a thickness approximately the same as the pipe wall thickness are recommended. Sidegrooved specimens will provide an acceptable lower bound J-resistance curve. However, certain theoretical limitations and practical complications exist in developing J-resistance curves. First, the J-integral computational method has certain limits of applicability that are associated with the assumptions and conditions from which they were derived. These limitations are related to certain assumptions regarding the stress-strain conditions in the region near the crack tip and translate into restrictions on structural size and material strength and toughness parameters to ensure valid analyses. Specifically, the ω restriction defined in Eq. (A-2) and the plane strain condition

$$b > 25 \frac{J}{\sigma_f} , \tag{A-20}$$

where

b = a characteristic structural dimension, in the case of laboratory J-R curve specimens equal to the uncracked ligament dimension must be satisfied.

When satisfied, these conditions are sufficient to ensure that the J-integral analysis technique can be applied rigorously. The requirement in Eq. (A-2) that ω be much greater than 1 is somewhat indefinite. Generally, an ω value between 5 and 10 is considered adequate, and a value of 5 is considered acceptable for the analyses being discussed here. Standard compact tension or bend bar specimens cannot meet the above validity criteria for large crack extensions, which must often be considered in the fracture mechanics analysis. The most desirable method of resolving this difficulty is to use nonstandard specimens that allow valid determination of the J-resistance curve at large crack extensions, e.g., large plan dimension compact tension specimens. This technique is recommended when possible; however, it is recognized that difficulties arise in trying to fabricate such specimens from pipe material. For weld material these type of specimens can be fabricated in a fairly straightforward manner. For base material, however, it is difficult to fabricate a specimen with prestrains representative of the piping material.

Where valid data cannot be generated for large crack extensions, some method of estimating the ductile fracture resistance at large crack extensions is necessary. The following procedure is recommended for making such an extrapolation for use in the fracture analysis. First, the J-resistance curve from small specimen tests is plotted in J-T space out to its maximum value of valid crack extension, per Eqs. (A-1) and (A-2). The J-resistance curve may then be extrapolated up to a J level twice the highest J level where valid data are available using a straight line tangent to the small specimen J-resistance curve at its point of maximum valid crack extension. This extrapolation procedure, illustrated in Figure A-5, is based on evaluation of J-resistance curves generated from large plan dimension compact tension specimens with large amounts of stable crack extension. These results indicate that the suggested extrapolation procedure will give a conservative estimate of the material resistance to ductile fracture. The extrapolation approach described, however, is valid for power-law fitting of the J-R curve and not linear fits of the J-R curve. Because of potential nonconservatisms, extrapolation of straight line representations of the J-resistance curve beyond valid data is not allowed, and data of this form will have to be considered on a case-specific basis.

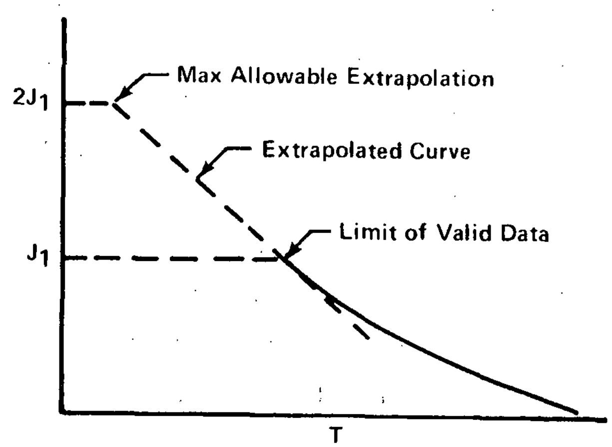

Figure A-5. Method for Extrapolating J-Resistance Curve in J-T Space

Table A-1 summarizes the recommended material properties tests to be conducted.

TABLE A-1. Suggested Material Properties Tests

Test Type	Temperature(a)	Number of Tests
	A. ARCHIVAL HEAT OF MATERIAL TEST MATRIX	
Tensile Test	High	3
	Low	1
J-R Curve	High	3
	Low	1

Test Type	Temperature(a)	Number of Heats	Number of Tests for Each Heat	Total Number of Tests
		B. NONARCHIVAL MATERIAL TEST MATRIX		
Tensile Test	High	3	2	6
	Low	1(b)	1	1
J-R Curve	High	3	2	6
	Low	1(b)	1	1

(a) High refers to a temperature near the upper range of normal plant operation.
Low refers to a temperature which may represent a plant condition (e.g., hot standby) where pipe rupture could have significant adverse consequences.

(b) Should be the same heat number as one of those tested at the high temperature.

A3.0 COMPARISON OF ANALYTICAL PREDICTIONS WITH EXPERIMENTAL RESULTS

To assess the accuracy of the computational methods described in Section A2.0, computations were performed to predict first crack extension and instability conditions for previously performed pipe experiments.* Predictions were made for two types of experiments, namely, 8-in.-diameter ferritic pipes[A.25] and two, 4- and 16-in.-diameter stainless steel pipe tests.[A.24] The pipe test section contained circumferentially oriented throughwall cracks ranging from about 20 to 30 percent of circumference in length. The pipes were subjected to pure bending moments and were instrumented to measure first extension of the initial crack and subsequent growth during the test. The stainless steel and ferritic pipe tests were performed at room temperature and 120 F, respectively. All tests were conducted to determine the value of moment and J at first crack extension, and maximum load; three of the ferritic tests were conducted to produce unstable crack extension. These aspects are presented in Section A3.2. Additional computations were performed by Battelle-Columbus to assess effects of pipe diameter and combined bending and pressure loads for stainless steel pipe. These results are presented in Section A3.3.

A3.1 ANALYSIS INPUT

In general, the analytical predictions were made using the three elastic plastic J estimation schemes outlined in Section A2.0 (Eq. A-7 and the NUREG procedure using Eqs. A-8 and A-11). Use of these equations requires the use of the tensile properties σ_o, E, α, n, and flow stress σ_f. These values were obtained for the test materials from actual stress-strain data obtained at the pipe test temperature.

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* Except as indicated in Section A3.3, computations utilizing the EPRI approach and the instability analyses of Section 3.4 were performed by Impell Corporation under NRC Contract No. NRC-03-84-070. Computations using the NRC staff analytical model and the model of NUREG/CR-3464 were performed by the NRC staff except that the staff used linear-elastic F factors developed by Battelle-Columbus for various R/t ratios rather than the simpler function, $F = 1 + 8(\frac{\theta}{\pi})^{5/2}$, as given in the NUREG document.

Prior to defining the appropriate Ramberg-Osgood parameters, a brief sensitivity study was performed to determine the strain that is applicable to the experimental condition. Based on this study, it was determined that strains of 1 percent and less comprised the region of interest for the ferritic pipe tests, while the appropriate strain for the stainless steel pipe test condition ranged from about 2 to 8 percent. The parameters that were used to fit the stress-strain data in these regions, and the flow stress are presented in Table A-2.

Table A-2. Material Properties Used in Analysis

Property	Material	
	Stainless Steel (Room Temperature)	Ferritic Steel (120 F)
α	1.91	1.35
n	4.7	6.2
σ_o - Reference Stress, ksi	30	35
E - Elastic Modulus, ksi	30×10^3	29×10^3
σ_f - Flow Stress, ksi	(a)	56.4

(a) 74.1 and 79.7 ksi for 4-in.- and 16-in.-diameter pipes, respectively.

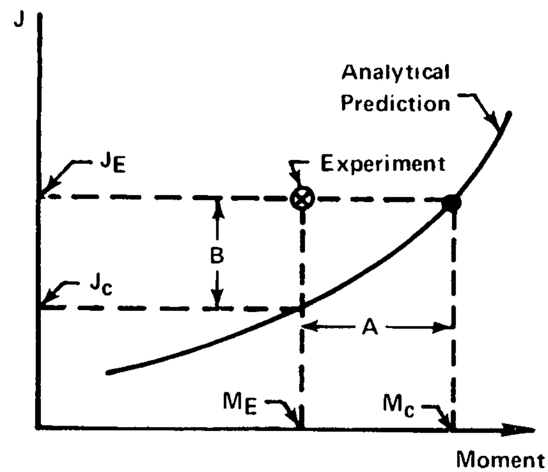
A3.2 FIRST CRACK EXTENSION PREDICTIONS

To compare the experimentally observed first crack extension conditions with the analytical predictions, J was calculated as a function of moment from Eqs. A-7, A-8, and A-11 for each of the pipe tests. A comparison with the experimental results was then made to determine the percent difference in the predicted to actual initiation moment at the observed value of J at initiation, and the ratio of calculated to experimental J at the observed value of moment at initiation. These differences are illustrated in Figure A-6, where

J_E and M_E are the respective J and moment values at initiation determined from the experimental pipe test results, and J_{IC} and M_C are respective calculated values of J at the observed initiation moment and moment at the observed initiation J. The percent difference relative to the experimental point are defined in Figure A-6.

The experimentally determined values of J_E and moment at initiation for the ferritic and stainless pipe tests are presented in Table A-3. The values of J calculated from the experiments (see Reference A.25) were determined using the following estimation scheme(A.8):

$$J = J_e + J_p = J_e + \beta \int_{\delta o}^{\delta} (2P)\,d\delta + \int_{\phi o}^{\phi} \gamma \, J d\Phi \qquad \text{(A-21)}$$



$$A = \frac{M_E - M_c}{M_E} \times 100 \text{ at Experimental Initiation J}$$

$B = J_c/J_E$ at Experimental Initiation M

Figure A-6. Predicted vs Experimental J and Moment from Ferritic and Stainless Steel Pipe Tests

Table A-3. Experimental Load Displacement Record From J and Moment Values at First Crack Extension for 8-in. Ferritic Pipe Test and 4- and 16-in. Stainless Steel Pipe Tests

Experiment Identification[a]	Observed Moment at Initiation (in.-lb)	Experimental J at Initiation (in.-lb/in.^{2})
N3 (A106 Grade B)	935,690	3680
N7 (A106 Grade B)	828,900	5400
N8 (A106 Grade B)	801,310	4420
N11 (A106 Grade B)	1,061,800	2340
N12 (A106 Grade B)	1,090,700	3110
N14 (A106 Grade B)	1,228,000	4300
N15 (A106 Grade B)	1,189,400	2850
B4 (Type 304 S.S.)	152,600	11,300
B16 (Type 304 S.S.)	6,609,000	20,600

(a) NX: N = U.S. NSRDC 8-in. ferritic pipe test, x = test specimen number
BX: B = Battelle stainless steel pipe test, x = nominal pipe diamater.

where β, 2P and γ are defined below Eq. A-14(b) and δ = platic load line deflection, ϕ = total crack angle. For this purpose, the calculated elastic displacements for the uncracked pipe were subtracted from the measured displacements, and it was assumed that the remaining displacement was due to the crack only.

The results from the comparison of the computations to the experimental results as defined in Figure A-6 are shown in Table A-4 for the ferritic and stainless steel pipe tests. From Figure A-6, positive values of percent difference in moment and factors greater than 1 for the ratio of J_{IC} to J_E indicate the computational results are conservative relative to the experimental results.

Table A-4. Comparison of Predicted and Observed Crack Initiation J and Moment for Ferritic and Stainless Steel Pipe Tests

	Percent Difference[a] in Moment to First Crack Extension at Initiation J			Ratio of Calculated J to Observed Initiation J at Initiation Load		
Experiment Identification[b]	EPRI (Eq. A-8)	NRC (Eq. A-11)	NUREG/ CR-3464 (Eq. A-9)	EPRI (Eq. A-8)	NRC (Eq. A-11)	NUREG/ CR-346 (Eq. A-
N3 (A106 Grade B)	17	-1	-4[c]	2.9	.95	.52
N7 (A106 Grade B)	10	-6[c]	-5[c]	1.8	.54	.52
N8 (A106 Grade B)	3	-17	-19[c]	1.1	.37	.26
N11 (A106 Grade B)	14	-3	-17	2.2	.84	.43
N12 (A106 Grade B)	16	1	-10[c]	2.6	1.1	.43
N14 (A106 Grade B)	5	-6	-16[c]	1.3	.63	.23
N15 (A106 Grade B)	16	4	-10[c]	2.9	1.4	.44
B4 (Type 304 S.S.)	8	-11[c]	-11[c]	1.5	.32	.10
B16 (Type 304 S.S.)	30	5	-6[c]	6.7	1.5	.32

(a) Experimental moment - Predicted moment x 100%.

(b) NX: N = U.S. NSRDC 8-in. ferritic pipe test, X = test specimen number
BX: B = Battelle stainless steel test, X = nominal pipe diameter.

(c) The calculated limit load was reached prior to the calculated J reaching the experimental J at initiation.

The information presented in Table A-4 shows that the EPRI estimation scheme is always conservative relative to the experimental observations, while the predictions from NUREG/CR-3464 are always nonconservative. As discussed in Section 5.0 of this report and earlier in this appendix, the NRC staff modified the analytical procedures of NUREG/CR-3464 to include the effects of strain-hardening of piping materials. Otherwise, the staff J analysis procedure is the same as in the NUREG document. While the staff modification improves the correlation between calculated and experimental results, they are still somewhat nonconservative except in a few cases. The absolute value of the maximum difference for the experimental moment for each computational method is approximately the same (i.e., about 20 percent), except for the 16-in.-diameter stainless steel pipe where the EPRI estimation scheme resulted in a 30-percent difference. In all but one case, both the staff and NUREG procedures predict that limit load will be reached prior to first crack extension. This depends on the selection of flow stress. The ratio of moment at initiation to the maximum moment observed from the experimental data ranged from about 0.90 to 0.99.

The predicted values of J for the ferritic pipe tests range from over-estimates by the EPRI and NRC modification of NUREG/CR-3464 methods by about factors of three to underestimates by NUREG/CR-3464 of about four. The predicted values of J for the stainless steel pipe tests range from overestimates by the EPRI method by about a factor of seven for the 16-in.-pipe to underestimates by factors of three to ten predicted for the 4-in.-pipe using the NRC and NUREG methods, respectively.

In addition to the information in Table A-4, comparisons were made to graph the difference in predicted and measured values of initiation J and moment relative to the limit moment and the scatter in the experimental data. These comparisons were made for the ferritic pipe tests (Cases N-3 through N-15) in Table A-3.

Figure A-7 illustrates the difference between the predicted and experimental values of J at the respective experimental initiation moments for each test. To include the experimental scatter, each J prediction was normalized by the average of the experimental initiation J values. These ratios were plotted against the ratio of experimental moment at initiation to

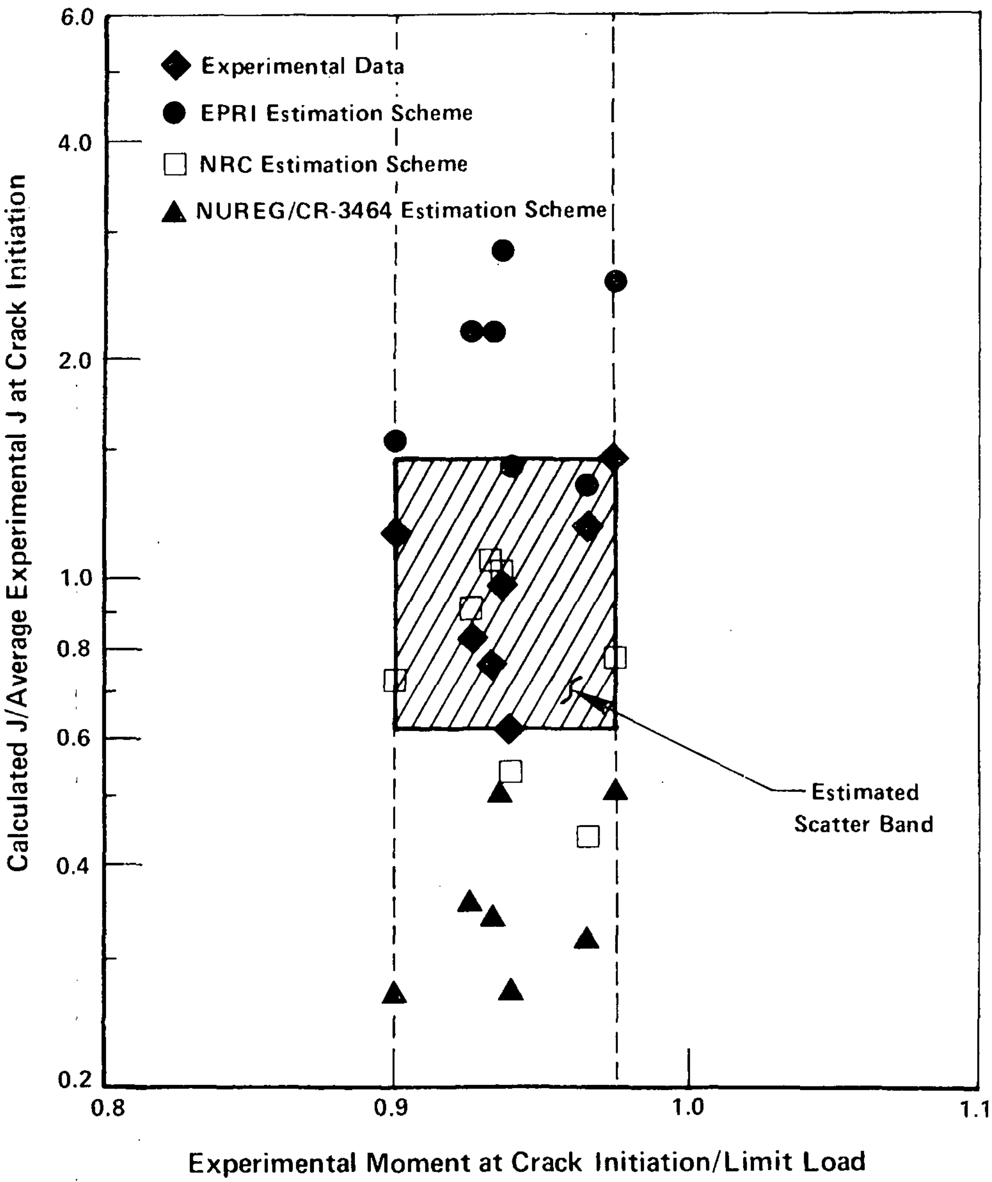


Figure A-7 Comparison of Predicted to Experimental J Values at Initiation for 8-in.-diameter Ferritic Piping Tests[A.25] experimental moment at limit load to show the proximity of first crack extension to observed limit load. The experimental scatter was estimated as a rectangle that enclosed all the experimental points. Any calculated J values from the estimation schemes that fall within the rectangle are considered to have negligible computational error. The data in Figure A-7 indicate that the NRC estimation scheme results most often lie in the scatter band while the results of the NURIG/CR-3464 estimation scheme are always on the nonconservative side of the band. The predictions from the EPRI estimation scheme are on the conservative side of the band for five of the seven tests, while the remaining two are within the band.

Figure A-8 shows the comparison for the initiation moment predicted from the estimation schemes, M, for each of the experimental initation J values, J_E. The predicted moments are normalized with respect to the observed limit moment for each respective test to show the predictions and experiments relative to limit load. From the information in Figure A-8, the EPRI estimation scheme provides conservative estimates of initiation moment for six of seven tests, while the NUREG/CR-3464 estimation scheme provides nonconservative estimates for initiation moment for six of seven tests. The NRC estimation scheme predicts initiation moments within the scatter ban for five of the seven tests.

Figure A-9 illustrates the calculated results of the three estimation procedures for Case N11. Point E is the experimental result. Interested readers may use their own computational techniques to derive J versus M results for this experiment.

Figure A-10 illustrates the effect of the assumed flow stress in the NUREG and the NRC modified NUREG procedures for Case B16, a 16-in.-diameter wrought stainless steel pipe. The higher limit load is based on a flow stress of 1.15 $(\sigma_u + \sigma_y)/2$ while the lower limit load is based on a flow stress of $(\sigma_u + \sigma_y)/2$; that is, without the factor of 1.15. Figure A-10 also illustrates Comment (c) under Table A-4. It is seen that the NUREG/CR-3464 procedure for this case results in reaching the limit load moment at a J value less than the experimental J at crack initiation. This was also the case for two of the NRC staff's analyses. The assumption made in these procedures, however, is that J tends toward large values which depend on the assumed kink angle after limit load is reached. Thus, if the analyst proceeds with an

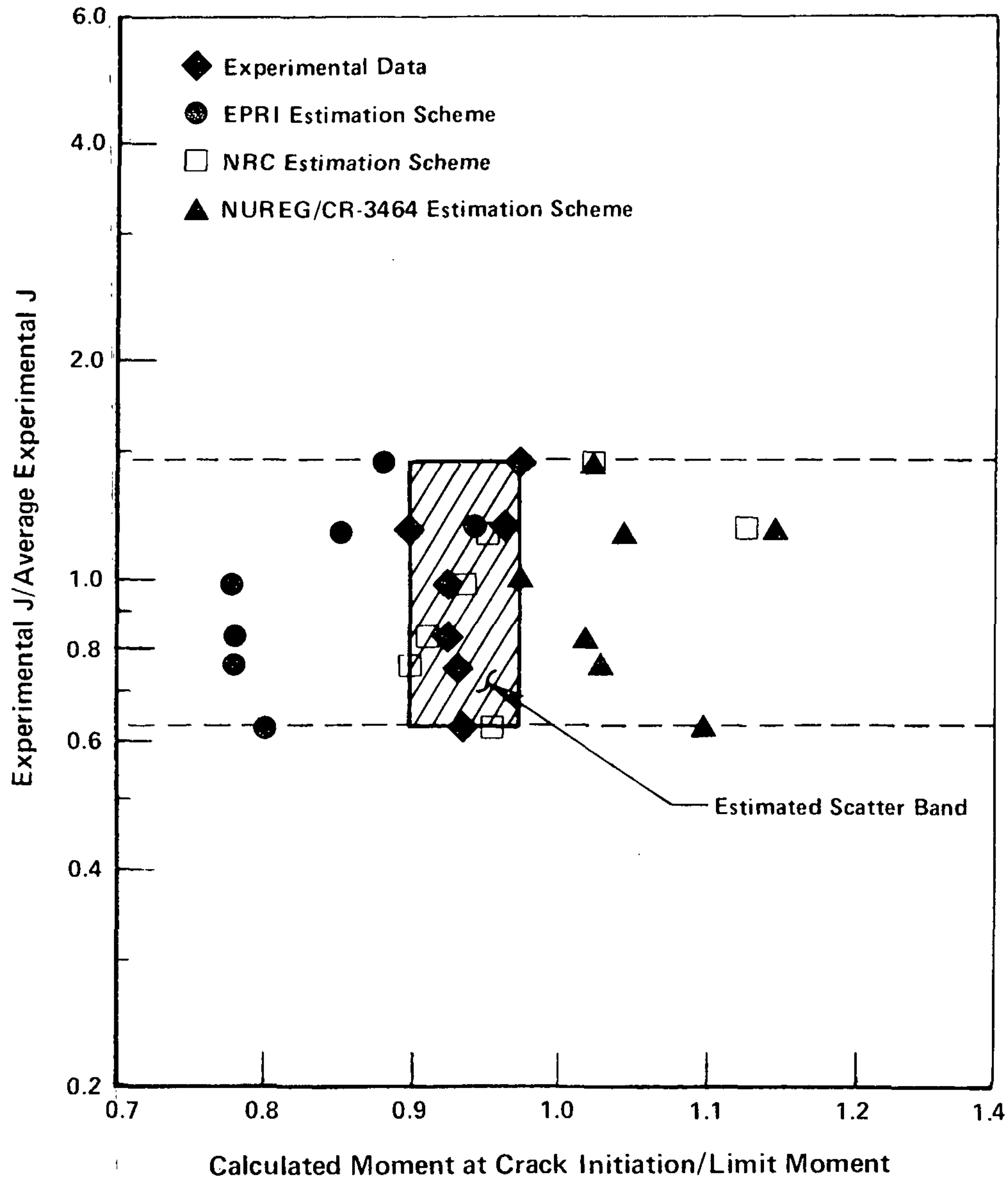


Figure A-8. Comparison of Predicted to Experimental Maximum Moments for 8-in.-diameter Ferretic Piping Tests (A.25)

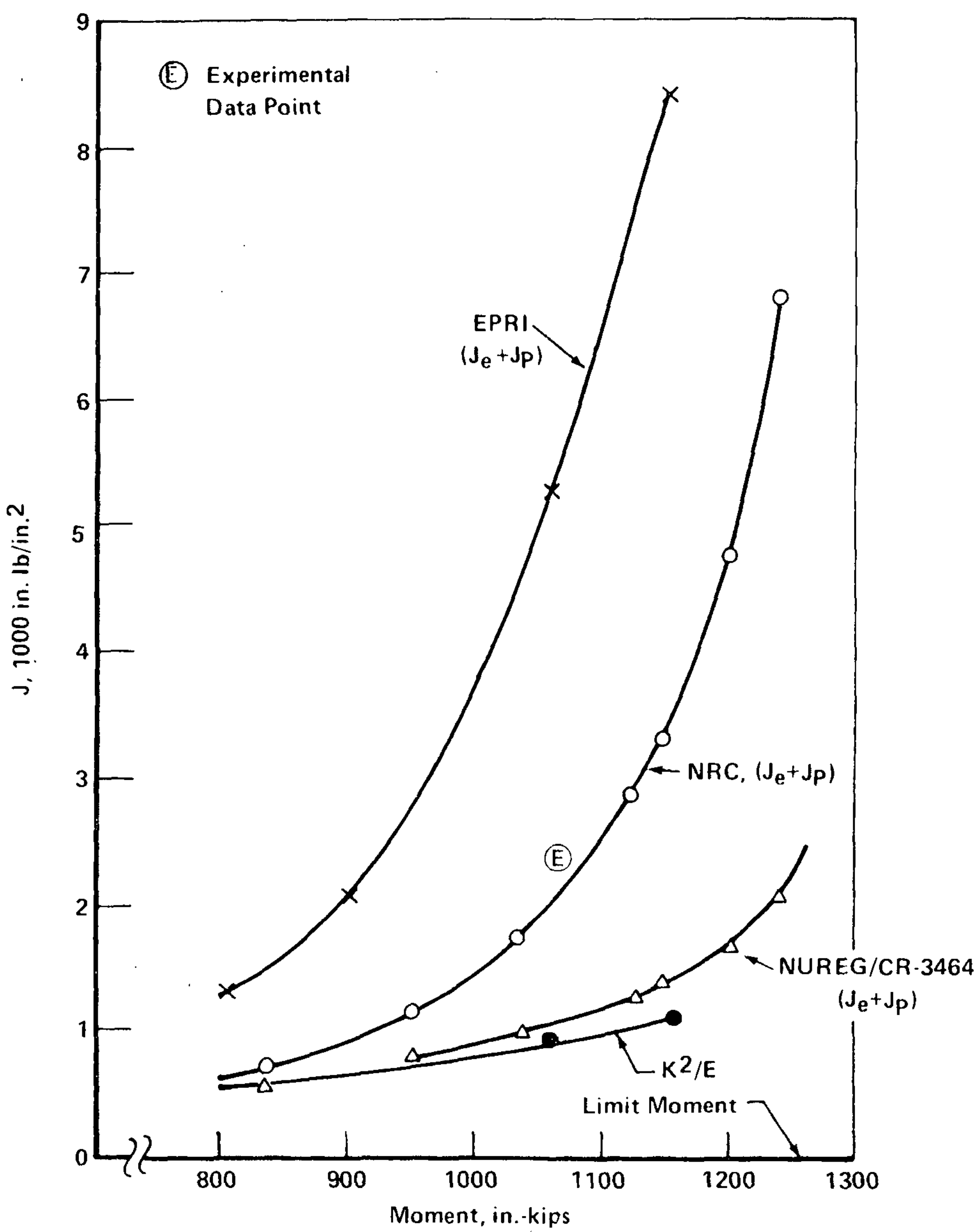


Figure A-9 Comparison of Various J-Estimation Schemes to Average Values From DTNSRDC Ferritic Pipe Test Data at Crack Initiation(A.25)

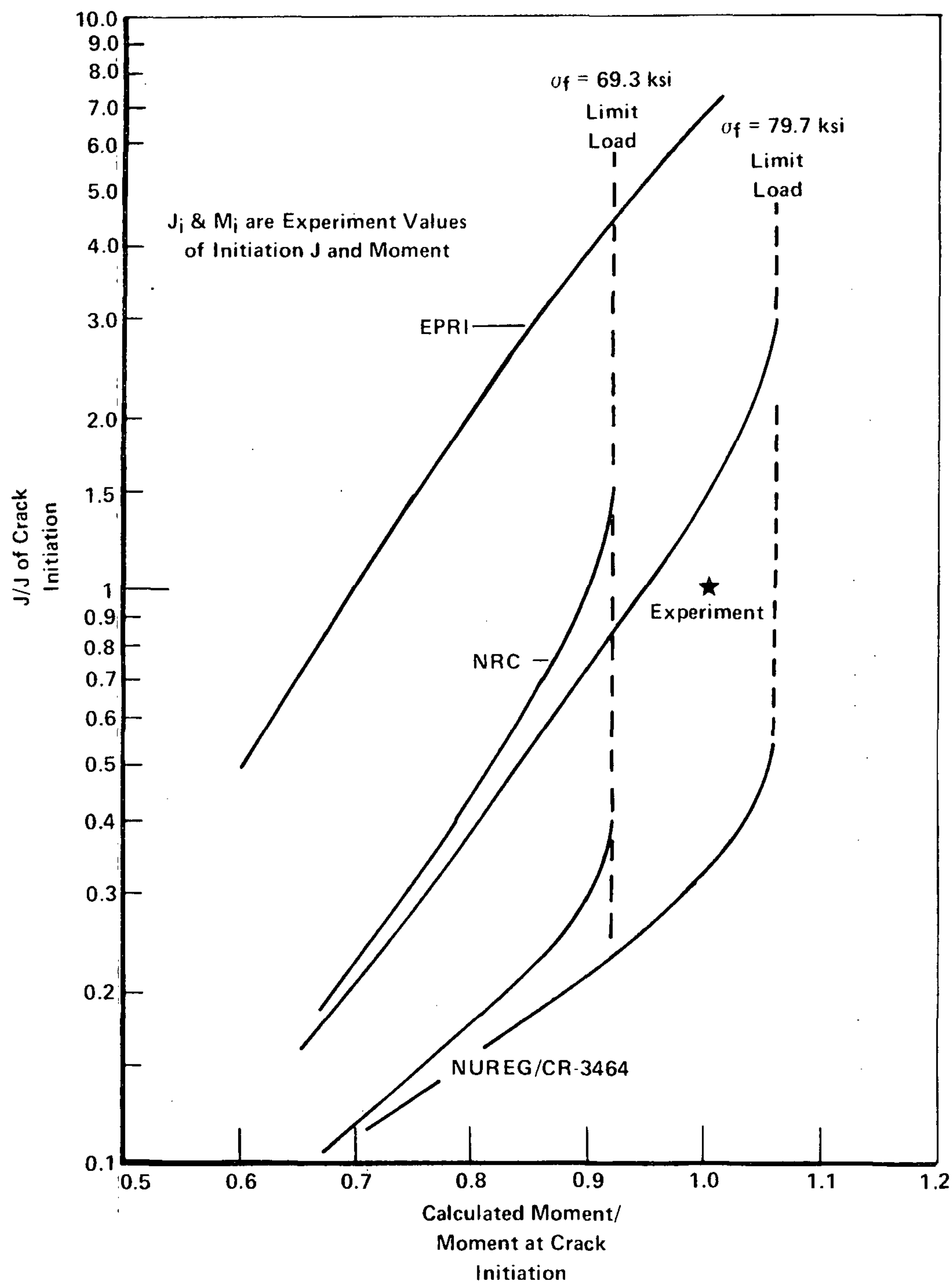


Figure A-10 Effect of Assumed Flow Stress on Predicted J and Moments for NRC and NUREG/CR-3464 Analyses. Experimental data point is 16-in.-diameter stainless steel pipe test conducted at Battelle.

instability analysis (see Section A3.4 which follows) for an actual application of this technology, conservative conclusions can still be reached. For example, refer to Figures 34 through 39 of Reference A.25.

A3.3 EFFECTS OF DIAMETER AND COMBINED LOAD ON PIPE EXPERIMENTS

The various J-estimation analyses have been assessed relative to existing throughwall circumferential cracked stainless steel pipe fracture experimental data, to see the effects of pipe size and loading conditions. Two specific evaluations are described in this section. The first evaluation involved comparing the various analyses for the case of pure bending (no axial tension) for different pipe diameters. The second set of calculations used combined bending and axial tension (pressure) pipe fracture data with a constant pipe diameter. All of the analyses reported in this subsection were performed by Battelle-Columbus.

A3.3.1 Diameter Effects Under Pure Bending

The comparisons here used past EPRI stainless steel pipe bending experimental data.[A.5] Table A-5 lists the experimental parameters. All pipe experiments involved total throughwall crack lengths of 37 percent of the pipe circumference. The nominal pipe diameters were 2, 4, and 16 in. The experimental load-displacement-crack length data were available to calculate the J and bending moment at crack initiation using an equation similar to Eq. (A-21).[A.22, A.26] For maximum load predictions, three-point bend bar J-R curves were used; however, only J-R curves for the 4-in.-diameter pipes were available at this time. Both the engineering and stress-strain curves were used for the analyses requiring Ramberg-Osgood relationships. Note also that for the stainless steels tested, the validity requirements were not met for either plane strain at crack initiation or crack growth, hence good agreement between the pipe tests and the estimation schemes should not necessarily be expected.

The calculated loads at crack initiation and maximum load relative to the experimental loads are given in Table A-6. One observation that can be made

Table A-5. Experimental Data for Type 304 Stainless Steel Pipes in Bending (With Throughwall Circumferential Cracks)

	Exp 7T	Exp 1T	Exp 8T
Outer diameter (D_O), in.	2.375	4.51	16
R (mean radius), in.	1.069	2.073	7.485
t (wall thickness), in.	0.237	0.354	1.030
R/t	4.51	5.85	7.26
$\frac{2a}{\pi D_O}$	0.371	0.371	0.3675
θ (half crack angle), degrees	66.78	66.78	66.15
Yield strength, ksi	36.4	38.6	45.8
Ultimate strength, ksi	87.4	90.2	92.8
Reduction of area, %	76.0	77.0	69.2
Net section stress at initiation, psi	74,864	70,537	75,604
Net section stress at maximum load, psi	75,823	71,775	78,811
Flow stress from tensile tests, psi 1.15 $(\sigma_y + \sigma_u)/2$	71,200	74,100	79,700

Table A-6. Stainless Steel Pure Bending Pipe Fracture Benchmark Calculations of Load at Crack Initiation and Maximum Load

Experimental Data						
Outside diameter, in.	2.375		4.50		16.0	
3 Point Bend Bar J_c, in.-lb/in.^{2}	3,000		5,000		13,000	
$2c/\pi D$	0.37		0.37		0.37	
Initiation Moment, in.-lb	29,620		152,600		6,609,000	
Maximum Moment, in.-lb	29,960		153,500		6,957,000	
Analytical Methods	Predicted/Experimental Loads					
	Init.	Max. Load	Init.	Max. Load	Init.	Max Load
G.E. Estimation Scheme using true σ-ε curve	0.81	--[a]	0.76		0.68	--[a]
using engin. σ-ε curve	0.91	--[a]	0.84		0.74	--[a]
NUREG/CR-3464 Analysis	0.81	0.81	0.94	0.94	0.89	0.89
NRC Analysis using true σ-ε curve	0.81	0.81	0.87	--	0.70	--
using engin. σ-ε curve	0.81	0.81	0.94	0.94	0.91	--
Net Section Collapse Analysis	N.A.	0.95	N.A.	1.03	N.A.	1.01

(a) Only J_c at initiation is available at this time, hence maximum load calculations requiring a J-R curve could not be made.

is that the net section collapse analysis predicted the loads at crack initiation and maximum load closely. In these calculations the flow stress was taken as 1.15 $(\sigma_y + \sigma_u)/2$ and a correction factor for ovalization was used.(A.5) This is due to the material toughness being very high and the experimental crack initiation was very close to the maximum load, hence the application of the net section collapse analysis is valid. The NUREG/CR-3464 and NRC analyses essentially are curve fitting analyses that interpolate between linear elastic behavior and net section collapse behavior. For the NUREG/CR-3464 analysis (Eq. A-8) initiation coincided with the limit moment due to the high J_{Ic} values of the bend bar specimens relative to the calculated applied J. Here the limit moment was based on $\sigma_f = (\sigma_y + \sigma_u)/2$ and no corrections for the pipe ovalization were included. For the NRC method (Eq. A-11) the limit moment and crack initiation were the same for the 2-in.-diameter and the 4-in.-diameter pipe when using the engineering stress-strain curve. The 16-in. pipe size moment at crack initiation was less than the limit moment.

For the EPRI estimation scheme, the predicted moments at crack initiation are conservatively lower than the experimental data. A trend of increasing conservatism with increasing pipe diameter can be observed here. This is consistent with past G.E. sensitivity studies that showed that larger pipes will have crack initiation and maximum loads below net section collapse conditions. For the maximum load only the 4-in. pipe J-R curve data from bend bar specimens were available at this time. This prediction of maximum load was much closer to the experimentally observed value. It is anticipated that when the J-R curves for the 2-in.- and 16-in.-diameter pipe become available, a similar trend will exist.

A second comparison made was to calculate the J at crack initiation using the experimental data. The calculated J values from the pipe experiments at crack initiation, J_{Ic}, relative to the three-point bend bar J_{Ic} values are given in Table A-7. Several observations can be made from this table. First both the three-point bend bar and calculated pipe J values at crack initiation increase with increasing diameter. The NUREG/CR-3464 values at crack initiation were much lower than the three-point bend bar J_{Ic} values. This was why this method predicted the initiation and maximum loads would be the same. The J values determined from the experimental data and Eq. (A-21) were 1.4 to 2.0

Table A-7. Comparison of Calculated J_C Values at Crack Initiation From Stainless Steel Pipe Bending Experiments to Three-Point Bend Bar J_C Values

Experimental Data			
Outside diameter, in.	2.375	4.50	16.0
3-Point Bend Bar J_C, in.-lb/in.^{2}	3,000	5,000	13,000
Analytical Methods	Predicted/3 Point Bend Bar J @ Initiation		
η-factor using P- δ record	1.40	2.06	1.56
G.E. Estimation scheme using true σ-ε curve	6.77	4.90	8.69
using engin. σ-ε curve	5.17	2.72	5.62
NUREG/CR-3464	-(a)	0.33	0.59
NRC Analysis using true σ-ε curve	-(a)	2.50	4.92
using engin. σ-ε curve	-(a)	0.76	1.62

(a) Experimental initiation load greater than predicted maximum load, so J at crack initiation could not be calculated.

times the bend bar J_C values, which was the most consistent trend. The EPRI estimation scheme pipe J_C values were considerably higher than the three-point bend bar J_C values. Using the true σ-ε curve gave higher J_C values than using the engineering σ-ε curve.

A3.3.2 Benchmark Calculations for Combined Tension and Bending

These comparisons used past EPRI stainless steel circumferentially cracked pipe fracture experiments. The combined pressure and bending pipe data consisted only of pressure (inducing the axial tension stress) and the maximum load for the initial crack length. No crack growth data were recorded. These data are given in Table A-8. Experiments with similar crack lengths but under pure bending were also conducted on different, but very similar pipes as noted in Table A-8.

The calculated maximum loads involved using the engineering stress-strain curves and the three-point bend bar J-R curve from the 1T and 2T pipes. The EPRI estimation scheme (Eq. A-7) was used to calculate J, and combined loads were accounted for by linealizing between bending and axial tension (see Figure A-11).

The results are graphically given in Figures A-12(a) and A-12(b), which are slightly different ways of presenting these comparisons. Figure A-12(a) shows the predicted/experimental maximum moments versus the axial tension stress. Figure A-12(b) is similar, but normalizes the axial stress to the bending stress. The comparisons showed that the accuracy of the EPRI estimation scheme maximum load for the axial tension combined with bending is very close to the benchmark calculations for the pipes under pure bending. The existing data were, however, limited to low axial to bending stress ratios, hence validity of extrapolation to significantly higher ratios was not examined due to a lack of data. Figure A-12(a) shows these calculations were reasonable for axial stresses where the axial stress, P_m, was less than half the ASME code design stress (i.e., $S_m/2$).

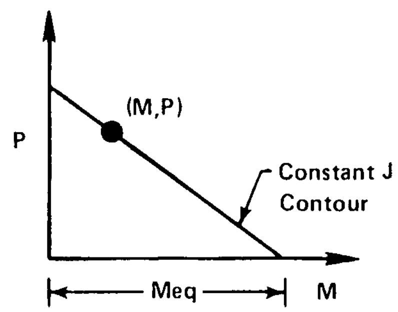
A3.4 INSTABILITY PREDICTIONS

Three of the ferritic pipe tests in Ref. A.25 were performed to produce crack instability. Computations were performed to determine the J and moment values corresponding to instability. Computations had already been performed for elastic-plastic and fully plastic displacement control loading.[A.25] Additional computations were performed by Impell Corporation using the EPRI J estimation scheme to determine the elastic-plastic, load-control loading instability point (see Eqs. A-7 and A-14a). The J computational schemes from NUREG/CR-3464 and its modified version (Eq. A-11) were not included in the

Table A-8. 4-in.-diameter Schedule 80 Type 304 Stainless Steel Pipe Data for Pure Bending and Combined Bending and Pressure

Test #	1T	585-1	2T	585-2
Outside Diameter, in.	4.5	4.5	4.5	4.5
Thickness	0.354	0.343	0.352	0.328
Yield strength, ksi	38.6	45.1	38.6	45.1
Ultimate strength, ksi	90.2	91.9	90.2	91.9
Percent elongation	79.6	69.0	79.6	69.0
$2c/\pi D$	0.371	0.371	0.229	0.229
Internal pressure, psi	0	1,050	0	2,500
Initiation Moment, in.-lb	152,600	--	235,460	--
Maximum moment, in.-lb	153,500	153,500	242,000	224,600

stability comparison because they did not include computational routines for tearing modulus, T. These three computations and the material resistance curve are illustrated in Figure A-1 as a J/T plot. The computations performed using the EPRI estimation scheme includes crack growth beyond J_{IC} as defined by the respective J-R Curves in Reference A-25. The comparison of the experimental J values of instability with the three prediction methods are presented in Table A-9.



$$Meq = M + \frac{R}{2}\,\frac{F_t}{Fb}\,P$$

Figure A-11. Combined Tension and Bending

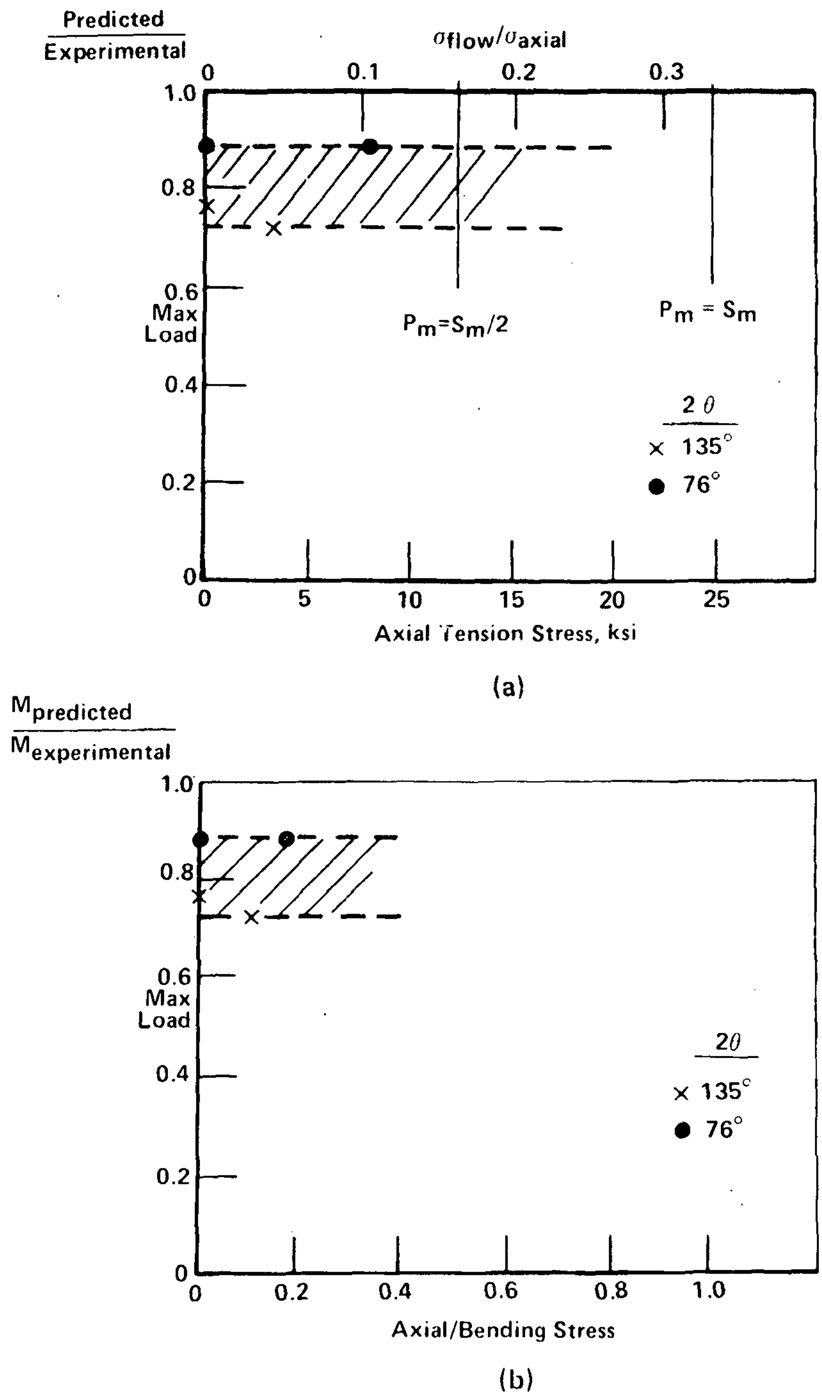


Figure A-12. Comparison of Predicted to Experimental Maximum Moments Using G.E. Estimation Scheme for 4-in.-diameter Schedule 80 Pipe Data

Table A-9. Comparison of Predicted vs Experimentally Determined Values of J at Instability

Experimental Identification[a]	J at Instability[b], in.-lb/in.^{2}			
	From Exp't Record[c]	EPDC[d]	EPRI Estimation Scheme[e]	FPDC[f]
N13	19,240[g]	10,600[g]	7,700[h]	6,300[h]
N14	18,730[g]	12,400[g]	9,200[h]	8,300[h]
N15	20,480[g]	12,800[g]	7,900[h]	6,100[h]

(a) NX: N = U.S. NSRDC Ferritic Pipe Test, X = Specimen number. All experiments were compliant displacement-controlled tests with instability after maximum load.

(b) All computations include crack growth.

(c) Using Eq. (A-21) and experimental load-displacement-crack growth data at instability as determined from Ref. A.25.

(d) EPDC = Elastic-Plastic displacement-control calculations from Ref. A.25.

(e) Using EPRI estimation scheme to predict load-controlled instability.

(f) FPDC = Fully plastic displacement control (Ref. A.7).

(g) Instability after max. load.

(h) Instability predicted prior to max. load.

The comparison of the experimental moment at instability with the computed instability moments are presented in Table A-10.

The results in Tables A-9 and A-10 show that the lowest value of J at instability for the three experiments is obtained from the fully plastic displacement control method (Ref. A.7), while the lowest moment at instability is predicted by the EPRI estimation scheme using assumed load control conditions. The EPRI estimation scheme predicts the lowest moment because for these experiments it computes a relatively high J value for moment near instability. This high J value will predict instability at lower moments.

Table A-10 Comparison of Predicted vs Experimentally Determined Moment at Instability

	Moment at Instability[b], in.-kip			
Experimental Identification[a]	From Exp't Record	EPDC[c]	EPRI Estimation Scheme[d]	FPDC[e]
N13	1,167[f]	1,197[f]	1,042[g]	1,198[g]
N14	1,316[f]	1,323[f]	1,273[g]	1,305[g]
N15	1,154[f]	1,207[f]	1,116[g]	1,215[g]

(a) NX: N = U.S. NSRDC Ferritic Pipe Test, X = Specimen number. All experiments were compliant displacement-controlled tests with instability after maximum load.

(b) All values include crack growth.

(c) EPDC = Elastic-Plastic displacement-control values from work in Ref. A.25.

(d) EPRI estimation scheme to predict load-controlled instability.

(e) FPDC = Fully plastic displacement control values from work in Ref. A.25.

(f) Instability after max. load.

(g) Instability predicted prior to experimental max. load.

A4.0 CONCLUSIONS AND RECOMMENDATIONS

- The J-R curves should be obtained using specimens whose thickness is equal or greater than that of the pipe wall. The specimen should be large enough to provide crack extension up to an amount consistent with J/T conditions determined by analysis for the application. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques may be used if appropriate as described in Section A2.4.

- The stress-strain curves should be obtained over the range from the proportional limit to maximum load.

- Ideally, the materials tests should be conducted using archival material for the pipe being evaluated. If archival material is not available, tests should be conducted using specimens from three heats of material having the same material specification. Test material should include base and weld metals.

- At least two stress-strain curves and two J-resistance curves should be developed for each of a minimum of three heats of material having the same material specification and thermal and fabrication histories as the in-service piping material. If the data are being developed from an archival heat of material, a minimum of three stress-strain curves and three J-resistance curves from that one heat of material is sufficient.

- The tests should be conducted at temperatures near the upper range of normal plant operation (e.g., 550 F). Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. One J-R curve and one stress-strain curve for one base metal and weld metal are considered adequate to determine temperature dependence.

- As indicated in Section 5.9.1 of the main text there are certain limitations that currently preclude generic use of limit load analysis to evaluate leak-before-break conditions for the purpose of eliminating pipe restraints. However, the task group believes that limit load analysis can be used to demonstrate acceptable leak-before-break margins for the application provided the limit moment is greater than the applied (normal operation plus SSE) moment at any location in the pipe run by a factor of at least three. Limit moment should be determined from Eq. (A-19) where the flow stress is determined from ASME Code properties. Data obtained from future tests (see Section 10.0) may provide information that would allow less restrictive use of limit load analysis for justifying elimination of pipe restraints.

- In an attempt to benchmark various J computational methods the study group compared various J analysis methods (see Section A2.3.1) with currently available experimental data that describe the moment and J values corresponding to first crack extension (see Table A-3) for ferritic and stainless steel piping. The results from this comparison (see Table A-4) inidicate that the EPRI estimation scheme is consistently conservative in perdicting moment to initiation with a maximum error of about 20 percent for ferritic piping and 30 percent for stainless steel piping. The method described in NUREG/CR-3464 was consistently nonconservative in predicting moment to initiation with a maximum error of about 10 percent for stainless steel and 20 percent for ferritic piping. The NRC modification of the NUREG predicted nonconservative results in the majority of cases with a maximum error of about 10 percent overprediction for stainless steel and 20 percent overprediction for ferritic steel.

- The EPRI estimation scheme consistently overpredicted the value of J at the experimental initiation moment. The computed J values differed by a maximum factor of seven for 16-in.-diameter stainless steel pipe and three for the ferritic pipe. The NUREG/CR-3464 estimation method consistently underpredicted the value of J at initiation. The computed J values differed by a maximum factor of ten for stainless steel pipe (4-in.-diameter) and four for ferritic pipe. The NRC modified NUREG method underpredicted J in the majority of cases. The computed J values were underpredicted by a maximum factor of three for both the stainless steel pipe (4-in.-diameter) and the ferritic pipe.

- Comparison to experimental data to assess the effects of pipe size showed that as the pipe size increased all the J-estimation analyses became more conservative.

- Comparison to experimental stainless steel pipe data to assess pure bending versus bending and axial tension showed that the degree of conservatism in the EPRI estimation scheme was the same for the two loading conditions. Only limited data were available in these comparisons and the axial stress were less than $S_m/2$.

- The guidelines developed for applying leak-before-break technology (see Section 5.0) are intended to provide adequate margin against full pipe break by selecting reasonably conservative analytical models, material properties, and margins on leak rate, load, and flaw size. However, analyses performed as part of this effort indicate that there can be significant differences between experimental results and predictions made by various computational procedures. These differences show that certain computational procedures are sometimes nonconservative; consequently, the analyst must take steps when applying the technology to ensure that nonconservative predictions are not made and the intended overall margins against full pipe break described in this report are maintained.

- When crack extension is predicted to occur, stability analysis should be performed (see Section 3.4 of the main text) to determine if adequate margin against crack instability are maintained. Stability computations should include crack extension characteristics of the materials as defined by appropriate J-R curve data.

- The experimental estimates for J_{IC} used to benchmark the computational methods (per Section A3.2) were based on actual pipe test results. Because J-R curves generally are not available from pipe sections, predictions of pipe integrity for specific licensing applications generally will have to be made based on other type specimen tests (e.g., compact tension or bend bar specimens). The data for the 8-inch-diameter ferritic pipe (Ref. A.25) indicate that the J-R curves obtained from pipe and compact tension specimens are essentially the same. However, the data in Tables A-3 and A-6 indicate that the J_{IC} for bend bars for stainless steel are significantly lower than that obtained from the pipe tests. These results indicate that using compact tension or similar specimens to make predictions for piping should be conservative for ferritic and stainless piping, provided they are obtained and applied as outlined in Section A2.4. Future pipe tests should provide additional information to quantify property differences between data obtained from pipe and other type specimens for a wider range of conditions.

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APPENDIX B

PROBABILISTIC FRACTURE MECHANICS METHODS

APPENDIX B

PROBABILISTIC FRACTURE MECHANICS METHODS

B1.1 INTRODUCTION

Over the past several years, probabilistic analysis techniques have gained increased acceptance as a method of evaluating the safety of nuclear power plants. One application has been through probabilistic risk assessment (PRA) of event sequences potentially leading to radioactive releases. A different application, which will be discussed here, probabilistically evaluates the adequacy of individual systems, structures, or components to resist failure when subjected to postulated design loads.

In essence, a typical component evaluation compares some measure of its strength -- material yield stress, for example -- against the stress resulting from anticipated loads applied to it. If strength exceeds stress, the component is considered adequate for the postulated loads. Should stress exceed strength, however, the component is presumed to fail.

As illustrated schematically in Figure B-1, a deterministic calculation compares point estimates of stress and strength to evaluate component adequacy. Generally, these are nominal values established according to conservative load limits and material strength parameters such as those defined by the ASME Code. In component design the application of "safety margins" provides an added measure of conservatism. The safety margin compensates for uncertainty associated with many factors, including:

- Variability in nominal material strength, that is, actual strength may be lower than that specified in the analysis.

- Degradation in material strength, such as embrittlement due to radiation.

- Variations in postulated loading conditions such as pressure and temperature transients.

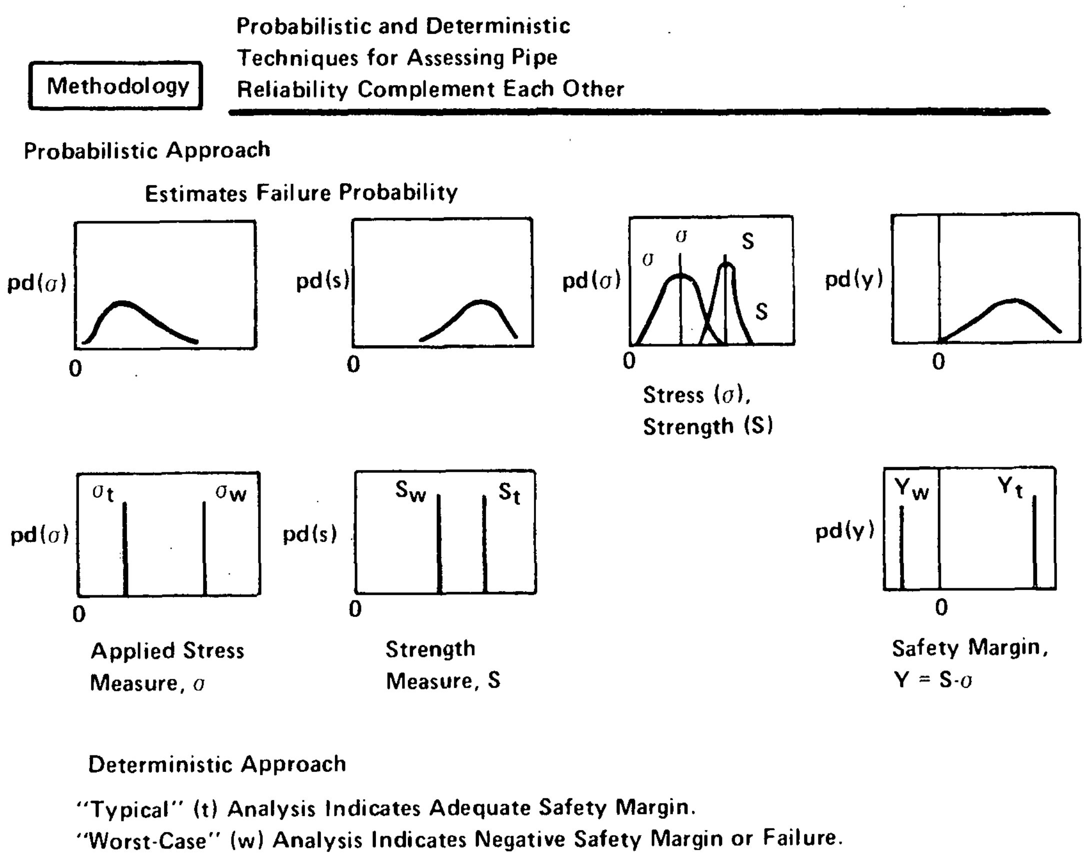

Figure B-1. Comparison of Probabilistic and Deterministic Techniques for Evaluating Failure

- Load conditions generally regarded as having secondary significance and which are therefore neglected in the evaluation.

- Unanticipated load conditions.

- Simplifications made in modeling a physical system.

- Approximation methods used to calculate stresses and resultant compnent response.

Stress and strength limits are generally set according to specific design considerations. It is not unusual that an evaluation based on "worst case" stress and strength values outside of the design scope will predict a negative safety margin, in other words, failure.

The deterministic approach embodies a significant degree of inherent conservatism. This conservatism stems from many sources as follows:

- The margin between code allowable limits and actual failure.

- The margin between design conditions and code limits.

- The particular analytical techniques used to predict component response to appplied loads.

- Input conditions used in predicting component response.

These conservatisms generally add together; thus, the more parameters involved, the more conservative a deterministic evaluation tends to be.

The probabilistic approach replaces the fixed values with random variables, each of which has a statistical distribution. Thus, variations in strength and stress about their nominal (or "best-estimate") values are explicitly considered. When plotted together (see Figure B-1), the area where these distributions overlap represents the probability that stress exceeds strength, in other words, that the component will fail. Instead of setting out to determine if a design is adequate and by what safety margin, a probabilistic evaluation estimates the failure probability ("reliability") of the design. The design is considered adequate ("safe") if the failure probability is acceptably low. What constitutes "acceptably low" is subject to judgment, usually taking into account the potential consequences of failure; the more serious the consequences, the lower the tolerable failure probability.

By distributing each parameter statistically, a probabilistic analysis yields results that more closely reflect reality. Moreover, probabilistic techniques can take event occurrence rate into account, and thus more realistically weight the relative effects of frequent vs infrequent load events on overall reliability. Statistical uncertainties attached to each distribution can be carried through the analysis to estimate the uncertainty in predicting reliability.

Because the simultaneous interaction of many individual--and often deterministically unrelated--factors is reflected in a single result (i.e., failure probability), probabilistic techniques provide a convenient yet powerful basis for sensitivity studies. For example, the relative contributions to piping reliability of material properties (strength, crack growth behavior) and nondestructive examination (inspection interval, crack nondetection probability as a function of depth) can be evaluated.

B1.2 PROBABILISTIC FRACTURE MECHANICS MODEL

The evaluations of double-ended guillotine break (DEGB) in reactor coolant piping performed by the Lawrence Livermore National Laboratory (LLNL) represent one application of probabilistic fracture mechanics to the subject of pipe failure (see Section 3.4 of the main text). In these evaluations, the probability of pipe break resulting from crack growth at welded joints ("direct" DEGB) is estimated using the procedure illustrated schematically in Figure B-2. The left column represents the analytical procedure, the right column the input information and analytical models used at each step of the

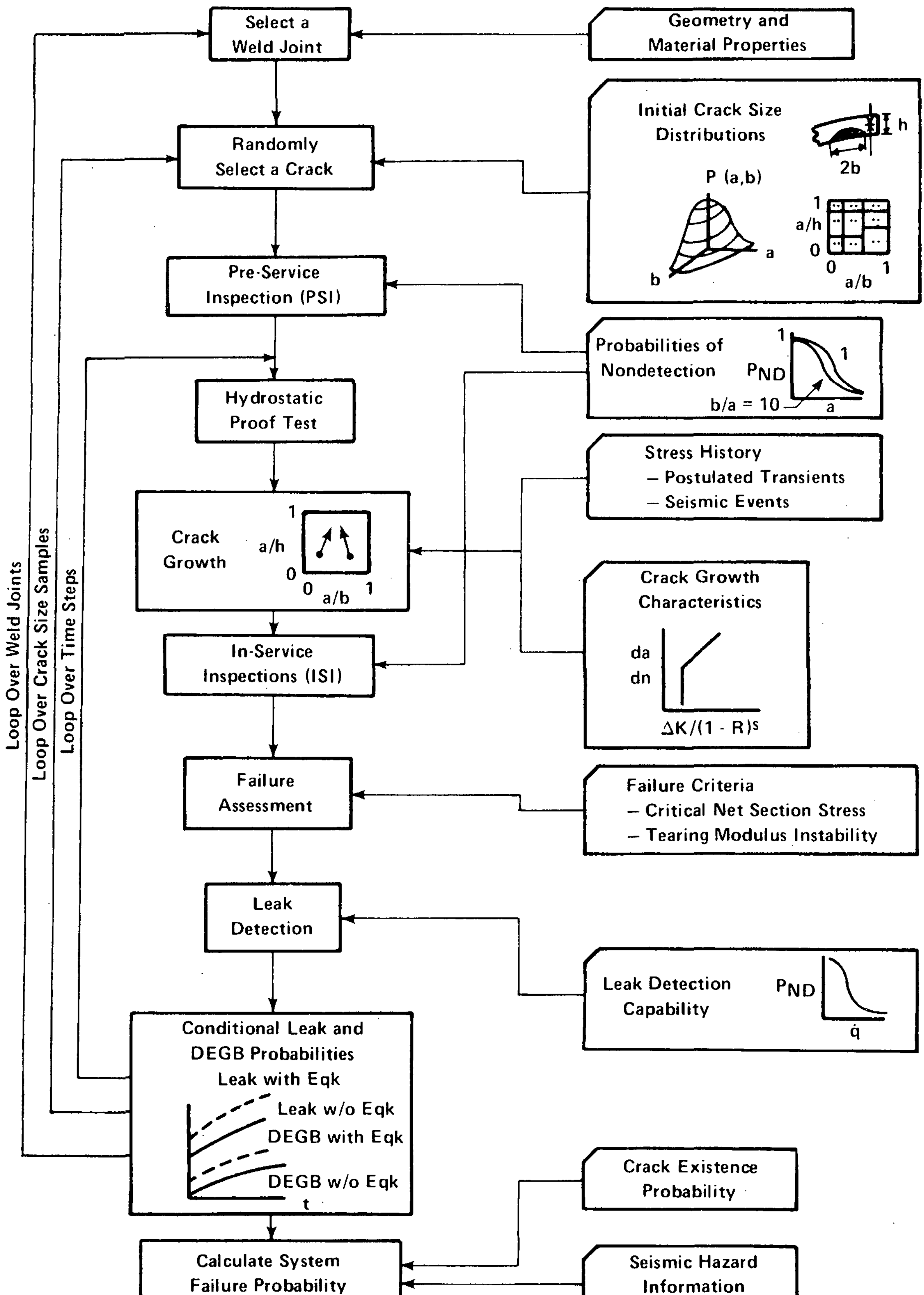

Figure B-2. Probabilistic Fracture Mechanics Procedure Utilized by LLNL in Evaluations of Reactor simulation. The procedure, described in detail by References B.1 and B.2, is summarized in the following discussion.

For each weld joint of a piping system, the leak or break probability is calculated using a Monte Carlo simulation technique. Each replication of the simulation--a typical simulation includes several thousand replications--begins with a preexisting flaw having initial length and depth randomly selected from appropriate distributions. These distributions in turn relate the conditional probability of crack existence. Fatigue crack growth is then calculated using a Paris growth law model, to which are applied stresses associated with normal operating conditions and postulated seismic events. The influence of such factors as nondestructive examination (NDE) and leak detection on failure probabilities is also considered through the inclusion of appropriate statistical distributions (e.g., probability of crack non-detection as a function of crack depth). Leak occurs when a crack grows through the pipe wall; break when failure criteria based on net section collapse or tearing instabilty are exceeded.

Completing all replications for a single weld joint and tabulating those cracks that cause failure yields the failure probability as a function of time at that weld, conditioned on a crack existing at the joint and an earthquake of given ground acceleration occurring. By combining the results for all welds in a particular pipe system, and then performing a systems analysis incorporating crack existence probability (a function of the total volume of weld material) and seismic hazard (which relates the occurrence rates of earthquakes as a function of peak ground acceleration), the non-conditional probabilities of leak or DEGB are obtained.

It is important to emphasize that this procedure is <u>not</u> a PRA-utilizing event and fault tree analysis. Instead, the procedure incorporates deterministic (either analytic or empirical) models into a probabilistic "framework" that allows the results of deterministic growth calculations for literally thousands of individual cracks to be consolidated, along with the effects of other factors such as NDE intervals and earthquake occurrence rates, into a single convenient result, namely the failure probability of a particular pipe system. This result could, in turn, provide input for that part of a PRA event tree using the probability of pipe system failure.

REFERENCES

B.1 D. O. Harris, E. Y. Lim, D. D. Dedhia, and H. H. Woo. June 1982. "Fracture Mechanics Models Developed for Piping Reliability Assessment in Light Water Reactors". Lawrence Livermore National Laboratory, Report UCRL-15490, NUREG/CR-2301.

B.2 T. Lo, H. H. Woo, G. S. Holman, and C. K. Chou. April 1984. "Failure Probability of PWR Reactor Coolant Loop Piping". Lawrence Livermore National Laboratory, Report UCRL. Presented at the ASME Pressure Vessel and Piping Conference, San Antonio, Texas, June 17-21, 1984.

APPENDIX C

APPENDIX C

This appendix relates to information presented in Section 7.0. The Pipe Break Task Group recognizes that the two major references are not readily accessible; therefore, these references both reproduced with approval of the authors make up the appendix as follows:

C-1- Industry Initiatives - Atomic Industrial Forum; Submitted by Patrick Higgins, AIF.

C-2- R. P. Schmitz, "Proposed Changes in Intermediate Pipe Break Criteria".

C-1 INDUSTRY INITIATIVES - ATOMIC INDUSTRIAL FORUM

The Atomic Industrial Forum's Committee on Reactor Licensing and Safety, through its Subcommittee on Load Combinations, has been active in the area of pipe break and load combinations in the design of nuclear piping systems for many years. In 1978 and 1980, the AIF had an exchange of correspondence with the NRC staff on the subject of load combinations in the design of nuclear piping systems. That correspondence also included discussion regarding the need to take a more rational approach to the general question of pipe break. The AIF Subcommittee at that time was aware of the substantial amount of work being done by both NRC and industry to further improve understanding of the probability of pipe breaks and on the testability of defects in piping well before such defects became critical. In early 1983 the AIF Subcommittee reviewed the status of that work and concluded that it was timely for regulatory action to provide more safety and cost-effective criteria for the design of nuclear piping systems. Accordingly, our recommendation on those matters were provided to the NRC in correspondence dated March 28, 1983. That correspondence encouraged the NRC to rationalize pipe break criteria applied to nuclear piping systems, and specifically encouraged that the guillotine break be removed as a design condition for the purpose of consideration of pipe whip. The March 28, 1983 correspondence provided proposed pipe break criteria to replace existing criteria.

Following receipt of a letter from NRC which largely agreed with the discussion of our March 28, 1983 correspondence, and encouraged further meetings, we met with the NRC staff to discuss our proposed pipe break criteria to replace existing criteria. The criteria was then refined over a period of months and transmitted to the NRC in correspondence dated July 14, 1983. That correspondence essentially called for elimination of the double-ended rupture in the primary coolant system as a design consideration for pipe whip and elimination of arbitrary intermediate breaks, where justification can be provided using state-of-the-art techniques. The criteria specifically calls for retention of a break area equivalent to an assumed double-ended pipe break for design of the emergency core coolant system, containment systems, and equipment qualification. Thus, the proposed pipe break criteria revisions were aimed primarily at elimination of pipe whip restraints, which in light of new information on probability of pipe break and detectability of flaws, were judged to be neither cost nor safety effective. In August of 1983, NRC responded and encouraged continued communication between AIF and the NRC staff to continue to develop and refine new pipe break criteria to allow elimination of the double-ended rupture in nuclear piping systems. The AIF work was undertaken largely because of the cooperation of the NRC staff and its management. It was viewed as a unique opportunity to increase nuclear safety while at the same time reduce plant costs.

VALUE-IMPACT

Discussion

Since mid-1983, the NRC has received numerous letters, not only from joint industry groups such as the AIF, but from various utilities, requesting that the NRC allow the elimination of specific double-ended pipe ruptures from the design bases of their respective plants. Typically, in these requests, the utilities attempted to provide an estimation of what they believe the value-impact to be. Value-impacts reported by industry to date identify significant cost savings and operational radiation exposure (ORE) reductions that can be realized based on NRC approval of elimination of pipe ruptures postulated at specific locations and subsequent elimination of the associated pipe ruptures mitigation hardware, i.e., pipe whip restraints (PWR) and jet barriers (JB).

Experience has shown that it is very difficult to assess an exact value impact. No standard value-impact methodology has been used due to the many variables involved in computing cost savings and ORE reductions. However, it is industry's belief that the magnitude of cost savings and ORE reductions are so compelling and beneficial that there is no need to rigorously quantify it.

The AIF reported in a March 28, 1983 letter to Harold Denton that the estimated total cost for design, procurement and construction of pipe rupture hardware is on the order of $20-$40 million per unit. This includes the cost for about 250 to 400 PWRs and 150,000 to 250,000 man-hours. This estimate is intended to represent an upper bound on possible savings available to a typical current vintage pressurized water reactor. For example, this estimate would represent the savings to a new plant, i.e., a plant that is just beginning the licensing process. However, as engineering, construction, and operation activities proceed to completion, the percentage of the total cost that can be saved decreases.

Where utilities have made specific requests, cost savings on the order of several million dollars and ORE reductions of several hundred man-rem are reported possible. The difference between the AIF and utility estimates is due to the fact that the AIF estimate is based on the total cost of all required pipe rupture hardware while the utility estimates are based on cost savings on eliminating hardware associated with a limited scope NRC request.

As can be expected, the magnitude of estimated savings differs from plant to plant, depending on the variables associated with a specific request. Some of the variables that become important in value impact estimates are related to (1) design; (2) plant status; (3) computational assumptions; and (4) scope.

Variables

Design - Value impact estimates can vary considerably from plant to plant due to fundamental differences in design. For instance, value impact estimates for a Combustion Engineering plant and a Westinghouse plant with similar megawatt ratings can be expected to be different due to design variances. Likewise, value impact differences can be expected among plants with the same NSSS vendor, e.g., 2-, 3-, and 4-loop Westinghouse plants. In addition, depending upon which lead A/E is contracted for the plant, design differences lead to different value impacts.

For A/Es, the major difference is layout. Further differences can originate as a result of design decisions made by the utility.

Plant Status - Plant status variables are important when considering value impact estimates. For example, a value impact estimate for an NTOL would include proportionately large contributions due to engineering and construction cost savings. A value impact estimate for an operating plant would include proportionately large contributions for removal costs and penalties for radiation exposure incurred during the removal.

Computational Assumptions - There are many computational assumptions used in value impact statements. Some examples are: magnitude of contractor fee for radiation exposure, cost per ton of PWR and JB steel installed, man-hours required for a given task, etc.

Scope - Scope breakdowns that enter into utility estimates are based on the following break categories:

1. A-2 breaks - under this activity associated with resolution of Unresolved Safety Issue A-2, which deals with asymmetric blowdown of

reactor pressure vessel due to rupture of reactor coolant system (RCS) piping at the reactor pressure vessel nozzles, A-2 breaks are considered to be those nozzle breaks at the vessel.

2. RCS breaks - these are the breaks postulated at the standard locations throughout the RCS, e.g., RCP suction.

3. Class 1 breaks - these are the breaks postulated in ASME III, Class 1 lines. These lines include the branch lines connected to the RCS.

4. AIB - arbitrary intermediate breaks (AIB) are the breaks that occur at locations in piping systems where the MEB 3-1 break criteria are not exceeded but where arbitrary locations are specified to meet the criteria for the minimum number of breaks.

The scope of value impact estimates can include savings for PWRs associated with these breaks or PWRs <u>and</u> JBs associated with these breaks. The two estimates could be significantly different.

<u>Cost Savings</u>

Cost savings reported by the industry can vary widely depending on which variables are considered in their respective request as noted above. All cost savings can be considered to fall into at least one or a combination of cost savings categories. These categories are engineering, construction and operational cost savings. Each of these general categories can be broken down into the sub-categories described below:

1. Engineering Cost Savings - this category can be considered to comprise (a) design costs which can be saved if pipe rupture mitigation hardware, i.e., pipe whip restraints (PWR) and jet barriers (JB), is not required due to pipe rupture elimination, and (b) analysis costs which can be saved if pressure, temperature, other environmental factors and system response do not have to be considered.

2. Construction Cost Savings - This category can be considered to comprise procurement, fabrication and installation costs for hardware which can be saved if PWRs and JBs do not have to be provided.

3. Operational Cost Saving - This category can be considered to comprise maintenance, inspection, accessibility and operational exposure costs, all of which can be saved if PWRs and JBs do not have to be provided. For example, maintenance and inspection costs associated with periodic surveillance of PWRs and JBs over the life of the plant can be eliminated. In addition, with improved accessibility, costs associated with worker inefficiencies and contractor fees for ORE can be reduced. The cost of replacement power for OLs is usually not included in the estimates.

Occupational Radiation Exposure Reductions

Industry request usually address occupational exposure savings in man-rem. The presence of PWRs and JBs contribute to increased occupational radiation exposures which are due mainly to maintenance and inspection requirements and/or inaccessibility. Reported savings also vary widely, based on the scope and assumptions used in the occupational exposure reduction value impact.

Conclusion

In conclusion, it is difficult to compare the reported value impacts due to the many variables involved in such estimations. However, it appears certain that there is sufficient basis to expect significant cost savings and ORE reductions no matter which variables or which scopes are considered. Therefore, when the costs and benefits are compared, the overall value impact of these industry requests is positive.

APPENDIX C-2

PROPOSED CHANGES IN INTERMEDIATE PIPE BREAK CRITERIA

For Presentation September 2, 1983

Committee on the Safety of Nuclear Installations
Meeting on Leak-Before-Break in Nuclear Reactor Piping Systems

by R. P. Schmitz, Chief Nuclear Engineer
Bechtel Power Corporation

Progress is being made in making the overall U.S. criteria for nuclear plants more rational. Hopefully, this will lead to better and safer plants for the future. These improvements are resulting from the increased use and understanding of risk assessment techniques and safety goals, as well as evaluation of the impressive operating experience being accumulated.

An important part of this effort is the review of criteria for nuclear plant piping systems and the development of more realistic safety-effective and cost-effective criteria for design. Our organization is giving this subject a very high priority. Improvements in pipe break criteria are a key part of this effort.

Pipe breaks have always been considered to some degree in commercial nuclear power stations in the U.S. At first, there was consideration only of potential radioactive releases. Next, emergency core cooling systems were added to replace the primary system coolant lost through the break. Later, criteria were developed for pipe whip restraints to protect against pipe movement. A detailed definition of break locations was required. Every year we added some detail to the definition of pipe breaks and their effects, including jet impingement loads, compartment pressurization, asymmetric loading on the reactor vessel, pump overspeed, effects on equipment supports, pipe dynamic impact loads, potential effects of pipes impacting smaller or larger pipes, formation of secondary missiles and formation of plastic hinges in the ruptured pipe. These effects are constantly being evaluated in greater and greater detail and presumably with greater accuracy. This progression has resulted from the tendency of engineers to achieve perfection, along with the reaction of engineers to the legalistic and adversarial atmosphere surrounding many projects. The regulators have encouraged this entire process, but industry must assume responsibility also.

The impact of these developments is just now being fully appreciated. A typical PWR now can have about 300 pipe whip restraints. The engineering effort on the part of the architect engineer required to deal with the entire problem can range up to 250,000 person-hours, more than was required for the entire balance-of-plant design work for many operating 500-600 MWe nuclear plants. Estimated costs for the design and construction work associated with pipe break effects for a typical unit are 30 to 50 million dollars. The design features included to protect against pipe whip clearly complicate the overall plant design, make access for maintenance and inservice inspection more difficult, and add to the dose accumulated by the plant operators for the life of the plant. These are real incentives to review, change and improve the pipe break criteria and practices now being used.

Regulatory criteria relating to piping design were essential for the design and construction of nuclear power plants but were promulgated prior to having the experience, analyzed data and detailed knowledge of the impact of the criteria that we have today.

Detailed analyses were recently completed to resolve NRC Generic Issue A-2 on asymmetric loads on the reactor vessel resulting from PWR main coolant pipe ruptures near the reactor vessel. Work by Westinghouse (WCAP No. 9570 - "Mechanistic Fracture Mechanics Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack"), Lawrence Livermore National Laboratory (LLNL) under contract to the NRC, and others has provided substantially convincing conclusions that, at least for the main cooling loop piping covered by these analyses, undetected defects that could cause guillotine and full size longitudinal breaks are incredible. The Lawrence Livermore work has also supported the argument that there is negligible safety benefit in combining pipe break and seismic loads. Combustion Engineering also participated in this review and recently formally requested changes on their docket for CESSAR Systems 80 to eliminate pipe breaks in the primary loop piping.

The NRC sponsored the LLNL work and closely monitored the work by others. They reviewed the results with the Advisory Committee on Reactor Safeguards subcommittee on March 29, 1983, and with the full committee on June 10, 1983. The NRC staff reported that they are technically satisfied with the arguments presented and that they plan to recommend appropriate changes to the current NRC criteria to eliminate mechanistic treatment of PWR primary loop breaks. They also plan to allow use of these criteria before the formal changes are implemented on a case-by-case basis.

Although the discussions to date relate specifically to the PWR primary loop, the technology and principles obviously apply to many other piping systems. There is a need to develop definitive criteria so that similar analyses can be made to attain the substantial benefits of this approach for other piping and locations.

These are important changes. However, there are a large number of other documents and criteria related to pipe breaks, such as containment design, emergency core cooling systems, equipment qualification, load combination equations, flooding, shielding, and jet impingement protection. The Atomic Industrial Forum's Subcommittee on Load Combinations is actively discussing proposed criteria changes for many of these subjects with the NRC. The NRC is considering forming a task force to make recommendations for all resulting changes. Hopefully, implementation of these changes can be completed in a year or less.

In order to attain a substantial and more immediate benefit, Bechtel proposed in a letter to the NRC on April 25, 1983 that the NRC eliminate from their criteria all intermediate breaks. The basic criteria for determining high energy line break locations are contained in Regulatory Guide 1.46 and Branch Technical Positions MEB 3-1 and ASB 3-1. These documents require that breaks be considered at terminal ends and at points where stresses or cumulative usage factors exceed specified limits <u>or at the two highest intermediate stress points</u>. On a typical PWR, breaks required at the two highest stress points represent more than half of the 300 break points, compared with 10 to 20 primary loop restraints.

We believe that current knowledge and experience indicate that assuming intermediate breaks at locations where stresses do not exceed ASME Code allowables is not justified and that, except for branch connections, this requirement should be deleted. There is now extensive operating experience with piping in over 80 operating U.S. plants and a number of additional similar plants overseas. We are not aware of any failure which indicates that designing for the intermediate breaks is necessary.

In addition, reason and logic indicate that postulating breaks based on the highest stress is not justified. These intermediate breaks are most often at locations where stresses are well below those susceptible to crack propagation. The present approach requires protecting against breaks at certain points but not at other points in the same system where stress levels may be only a few percent less. It also results in inconsistent approaches from system to system. In fact the number of breaks in branched piping systems depends more on the capability of the computer program used to handle all the branches in a single analysis than on the physical conditions occurring within the piping systems.

While the restraints associated with intermediate breaks represent more than half of the restraints, these restraints represent a disproportionately high percentage of the cost of the overall restraint design and installation effort. The location of terminal end breaks, and hence the location of their associated pipe whip restraints, is known as soon as piping layout and preliminary stress analyses are completed. This allows structural embeds to be located and placed before pouring concrete, space to be allocated for restraints and supporting steel, and safety-related targets to be routed away from the vicinity of the postulated break. The locations of intermediate breaks, on the other hand, are not known until the detailed piping and hanger design and subsequent stress analysis are completed. Even then, the addition of new piping system tie-ins or modifications to piping or hanger details due to field interferences and other reasons will often change the stress at different points in the line requiring changes in the location of intermediate breaks. The impact that changes of this nature have on the construction schedule during the latter stages of construction and startup is substantial.

Access during plant operation for maintenance and/or inservice inspection is hampered due to the congestion created by these restraints and the supporting structural steel, and due to the need to remove some restraints to gain access to welds. In addition to the increased work load, a significant increase in man-rem exposure is involved. Also, the need to verify adequate cold and hot clearances between pipe and restraint during initial heatup requires additional hold points during this already critical startup phase.

Recovery from unusual plant conditions would also be hampered by this congestion. In the event of a radioactive release or spill inside the plant, decontamination operations would be much less effective due to the complex shapes represented by the structural framework supporting the restraints. These effects would work to increase man-rem exposures associated with decontamination and restoration activities. Access for control of fires within these areas of the plant would be more difficult, especially under low visibility conditions. Substantial overall benefits in these areas would be realized by reducing the number of whip restraints required.

By design, whip restraints fit closely around the high energy piping with gaps typically on the order of half an inch. These restraints and their supporting steel significantly increase the heat loss to containment. Also, since thermal movement of the piping system during startup and shutdown could deform the piping insulation against the fixed whip restraint, the insulation must be cut back in these areas, creating convection gaps adjacent to the restraint, also increasing heat loss to containment. This effect is particularly pronounced with metal reflective insulation. The heat loss from 1 foot of uninsulated pipe is equivalent to the heat loss from approximately 200 feet of completely insulated pipe. Thus, the addition of whip restraints yielding a net increase in heat loss equivalent to 6 inches of uninsulated pipe per 100 feet of pipe would double the piping heat loss inside containment. This is a major contributor to the tendency of many containments to operate at temperatures very near technical specification limits. The elimination of whip restraints associated with intermediate breaks would assist in controlling the containment temperatures.

There is a small but finite possibility that installation, inspection or maintenance procedures involving whip restraints would not leave proper clearances between the restraints and the pipe, thus causing higher stresses in the pipe. Reducing the number of restraints decreases the chances of this happening.

Some consideration is being given to continuing the requirement for environmental qualification of equipment, protection against flooding and possibly some other effects of leaks in place of these intermediate breaks.

Overall, we are extremely pleased with the progress being made on improving the criteria for pipe breaks in light water plants. We have very actively supported this effort and will continue to work toward complete implementation of new criteria because we believe that substantially better future plants will be the result.

APPENDIX D

PARTICIPANTS IN TASK GROUP ON PIPE BREAK

APPENDIX D

PARTICIPANTS IN TASK GROUP ON PIPE BREAK

The following individuals participated in the initial plans for and the writing of NUREG-1061, Volume III, as members or consultants to the Task Group.

Members

R. W. Klecker, Chairman, NRC-NRR
S. H. Bush, Review & Synthesis Associates
S. H. Hou*, NRC-NRR
J. Strosnider, NRC-RES
K. Wichman, NRC-NRR

Consultants

C. K. Chou*, LLNL
R. Gamble, Impell Corporation
G. Holman, LLNL
G. Wilkowski, BMI-Columbus

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* Messrs. Hou and Chou participated actively in initial plans; however, they were unable to participate directly in the report preparation. Mr. Holman covered the activities of C. K. Chou during report preparation.

APPENDIX E

NRC MEMORANDUM INITIATING RULEMAKING

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 2 9 1984

MEMORANDUM FOR: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR INITIATION OF RULEMAKING REGARDING ALTERNATIVES TO POSTULATION OF PIPE BREAKS AND PROTECTION AGAINST ASSOCIATED DYNAMIC EFFECTS

The purpose of this memorandum is to request initiation of rulemaking to codify the use of advanced fracture mechanics technology in the regulatory process by restructuring pertinent parts of the regulations regarding the design basis for protection against the dynamic effects of a postulated pipe break. The Commission's regulations, as interpreted by OELD and currently implemented in the applicable Standard Review Plans and Regulatory Guides, impose the postulation of piping ruptures in high energy fluid systems, for both inside and outside of containment, as a part of the design bases for safety-related structures, systems and components.

Background

In 1975 a generic safety concern was identified that initiated Unresolved Safety Issue A-2. This issue involves the previously analyzed asymmetric blowdown loads that would be generated from postulated breaks in PWR reactor coolant main loop piping. Since that time the fracture mechanics technology regarding the potential rupture of tough piping such as used in LWR primary coolant systems, has advanced considerably. Both the NRC and the industry have spent significant time and effort to develop analytically and validate experimentally advanced fracture mechanics technologies applicable to pressure retaining components including piping systems.

These advanced fracture mechanics techniques deal with relatively small flaws in piping components (either postulated or real) and examine their behavior under various pipe loads. The objectives of these advanced fracture mechanics techniques is to demonstrate by analysis that the detection of small flaws either by inservice inspection or by leakage monitoring systems is assured long before the flaws can grow to critical or unstable sizes which could lead to large break areas such as the double-ended LOCA or its equivalent. The concept underlying such analyses is referred to as "leak-before-break."

Advanced fracture mechanics technology was applied recently in topical reports which were submitted to the staff by Westinghouse on behalf of 11 licensees belonging to the A-2 Owners Group. The topical reports for those licensee's plants were intended to resolve the issue of asymmetric blowdown loads on the PWR primary systems that resulted from a limited number of discrete break locations as stipulated in the resolution of USI A-2. However, the topical reports also demonstrated that the potential for failure of the main loop primary coolant piping for those plants is so low that protection against the dynamic effects of postulated breaks at any location in that piping need not be provided, thus eliminating the need for installation of pipe whip restraints or jet impingement shields.

After our evaluation of the Westinghouse topical reports, the staff developed a package for CRGR review which included (a) the staff's topical report evaluation containing justification for granting exemptions from GDC 4,1/ (b) the plan for implementation, and (c) the regulatory (value-impact) analysis.

Current Status

The NRC staff met with the CRGR to review this issue on September 28, 1983. In the minutes of that meeting dated October 14, 1983, the CRGR recommended that the EDO accept the staff's technical findings and proposed actions with respect to postulated asymmetric blowdown loads. The CRGR observed that these findings and the technical justifications in support of the findings could extend to other break locations and to assumptions previously made for piping loops and components of the reactor coolant systems, for piping connected to the coolant system and perhaps to the piping of other systems in the plant. To maximize the utility of the staff's recommendation and their potentially positive benefits to plants under construction, the CRGR recommended a special staff effort to implement these recommendations to the extent justifiable in terms of safety and staff resources.

Several PWR applicants with Westinghouse NSSS have submitted information to demonstrate the applicability of those Westinghouse topical reports. Combustion Engineering is seeking similar relief for its CESSAR facilities supported by its submittal of fracture mechanics analyses and materials

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1/ The justification for granting exemptions to GDC 4 was applicable to the protective measures (e.g., pipe whip restraints, jet impingement shields) against the dynamic loads associated with the definition of a LOCA as including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

data. These submittals are currently under review and additional submittals are expected shortly. OELD has provided the legal opinion that licensing actions regarding the Westinghouse A-2 package and any subsequent applications will require granting of exemptions to the regulations.

Requested Action

The granting of plant specific exemptions to the regulations on a system unique basis entails significant allocation of resources both by the NRC staff evaluating such requests for exemptions and by the industry performing appropriate analyses. In addition, OELD views are that extensive use of exemptions to authorize the elimination of pipe whip restraints is inappropriate. Accordingly, we are requesting the Office of Nuclear Regulatory Research to initiate rulemaking to enable the use of advanced fracture mechanics technology to determine the appropriate dynamic effects to be considered for piping system failures. There is a need for rulemaking that could allow less than full double-ended pipe breaks to be postulated for design against consequent dynamic effects (e.g., pipe whip, jet impingement). This rulemaking should not affect any other design basis requirements based on a double-ended pipe break, such as ECCS or containment loadings.

One way to accommodate these requested changes would be to modify General Design Criterion 4 to separately define the environmental and dynamic effects of postulated piping failures. Alternatively, additional guidance for determining appropriate dynamic effects for specific piping components could be provided in a new paragraph for 10 CFR. The need for additional guidance on postulated pipe breaks is noted in footnote 1 to 10 CFR 50, Appendix A.

This request has the concurrence of the CRGR as reflected in Minutes of CRGR Meeting Number 47, dated October 14, 1983.

Schedule

This task should be initiated as soon as practicable and proceed on an expeditious basis. We believe that this task is sufficiently urgent to warrant completion within one year.

Task Coordination

Technical direction should be conducted with full participation and concurrence of the NRR staff. B. D. Liaw, Chief, Materials Engineering Branch, Division of Engineering is designated as the NRR cognizant individual.

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

cc: W. J. Dircks
V. Stello

APPENDIX F

LIST OF ACRONYMS

APPENDIX F

LIST OF ACRONYMS

A/E	Architect Engineers
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
AIF	Atomic Industrial Forum
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASB	Auxiliary Systems Branch
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CRGR	Committee for Review of Generic Requirements
CRLS	Committee on Reactor Licensing and Safety
CSNI	Committee on the Safety of Nuclear Installations
CTOA	Crack Tip Opening Angle
DEGB	Double-Ended Guillotine Break
ECCS	Emergency Core Cooling System
EDO	Executive Director for Operations
EPRI	Electric Power Research Institute
FAD	Failure Assessment Diagram
FRG	Federal Republic of Germany
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HDR	Heissdampfreaktor or Superheated Steam Reactor
IGSCC	Intergranular Stress Corrosion Cracking
IPIRG	International Piping Integrity Research Group
ISA	Instrument Society of America
JB	Jet Barriers
LBB	Leak-Before-Break
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss-of-coolant accident
LWR	Light Water Reactor

MEB	Mechanical Engineering Branch
MHA	Maximum Hypothetical Accident
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Suppply System
NTOL	Near-Term Operating License
ORE	Occupational Radiation Exposure
PRA	Probablistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized Water Reactor; also Pipe Whip Restraint
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RSK	Reaktorsicherheitskommission (Reactor Safety Commission)
SEP	Systematic Evaluation Program
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake
USI	Unresolved Safety Issue

NRC FORM 335
(2-84)
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U.S. NUCLEAR REGULATORY COMMISSION

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13. ABSTRACT *(200 words or less)*

The Executive Director for Operations of the U.S. Nuclear Regulatory Commission (NRC) requested that a comprehensive review be made of NRC requiremeuts in the area of nucle power plant piping. In response to this request an NRC Piping Review Committee was formed. The activities of this review committee were divided into four tasks handled by appropriate task groups, namely: Pipe Crack Task Group, Seismic Design Task Group, Pipe Break Task Group, aud Dynamic Load/Load Combination Task Group. This report was prepared by the Pipe Break Task Group and deals with the potential for pipe breaks anc recommends modifications to the existing position. Specifically, this report contains the Task Group's recommendations for application of the leak-before-break (LBB) approa in the NRC licensing process. The LBB approach means the application of fracture mechanics technology to demonstrate that high energy fluid piping is very unlikely to experience doubled-ended ruptures or their equivalent as longitudinal or diagonal splits.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS
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